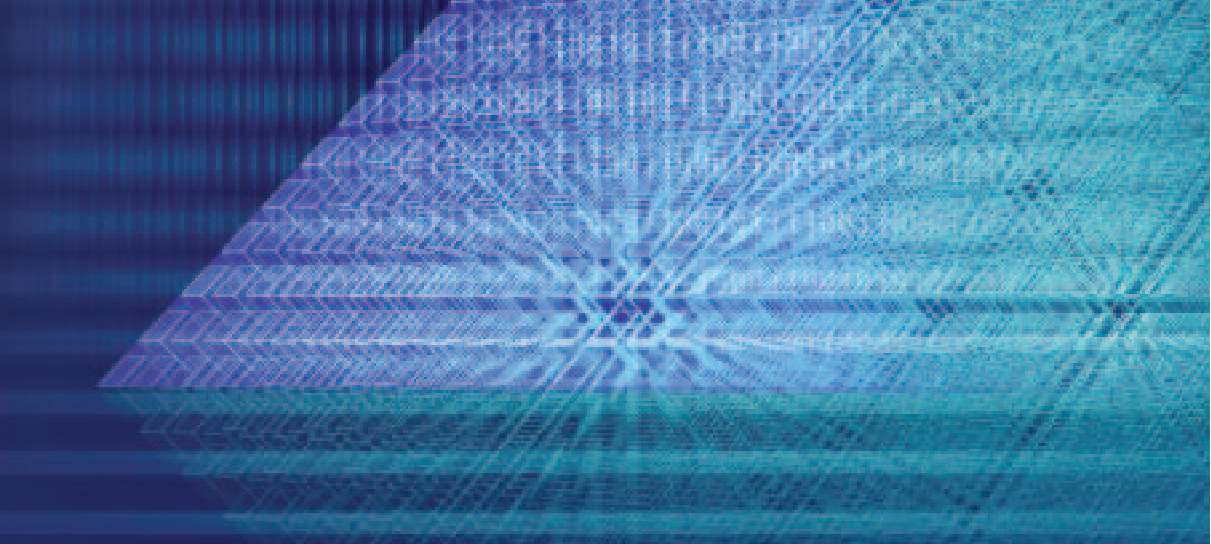

TECHNICAL REPORTS SERIES NO. 415

Status and Advances in MOX Fuel Technology

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 2003

STATUS AND ADVANCES IN MOX FUEL TECHNOLOGY

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BANGLADESH
BELARUS
BELGIUM
BENIN
BOLIVIA
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BULGARIA
BURKINA FASO
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHILE
CHINA
COLOMBIA
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ESTONIA
ETHIOPIA
FINLAND
FRANCE
GABON
GEORGIA
GERMANY
GHANA
GREECE
GUATEMALA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
LATVIA
LEBANON
LIBERIA
LIBYAN ARAB JAMAHIRIYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITIUS
MEXICO
MONACO
MONGOLIA
MOROCCO
MYANMAR
NAMIBIA
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORWAY
PAKISTAN
PANAMA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
SAUDI ARABIA
SENEGAL
SERBIA AND MONTENEGRO
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
THE FORMER YUGOSLAV REPUBLIC OF MACEDONIA
TUNISIA
TURKEY
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VENEZUELA
VIETNAM
YEMEN
ZAMBIA
ZIMBABWE

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

Printed by the IAEA in Austria
May 2003
STI/DOC/010/415

TECHNICAL REPORTS SERIES No. 415

STATUS AND ADVANCES IN MOX FUEL TECHNOLOGY

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2003

IAEA Library Cataloguing in Publication Data

Status and advances in MOX fuel technology. — Vienna : International Atomic Energy Agency, 2003.
p. ; 24 cm. — (Technical reports series, ISSN 0074–1914 ; no. 415)
STI/DOC/010/415
ISBN 92–0–103103–3
Includes bibliographical references.

1. Plutonium. 2. Nuclear fuels. 3. Spent reactor fuels. 4. Nuclear reactors. 5. Fast reactors. 6. Radioactive wastes. 7. Nuclear reactors — Decommissioning. 8. Hazardous wastes — Transportation. I. International Atomic Energy Agency. II. Series: Technical reports series (International Atomic Energy Agency) ; 415.

IAEAL 03–00317

FOREWORD

In May 1999, the IAEA, in cooperation with the OECD NEA, organized a Symposium on ‘MOX Fuel Cycle Technologies for Medium and Long Term Deployment’ in which more than 150 participants from 28 countries and four international organizations took part. In addition to dealing with current technologies for the MOX fuel cycle, the Symposium considered the future place of plutonium recycle. In this respect, it provided a comprehensive picture of the situation as of the end of December 1998. The Symposium was commended by the IAEA Board of Governors at their meeting in June 1999. It was proposed that a review be made of the status and development trends of advanced MOX fuel technologies based on the Symposium papers and on updates to account for developments since then. In addition, the long term initiatives which need to be put in place to ensure acceptance by the public of plutonium recycle, to offer a very long term environmentally friendly and sustainable energy supply, and at the same time ensure high security against proliferation, were to be examined.

This report represents an overview of the worldwide state of plutonium fuel development as of December 2000 with an outline of future trends. The review was prepared by a group of experts in the field, under the chairmanship of H. Bairiot, and supported by information from specialists in plutonium fuel developments and related subjects that the IAEA engaged. Information on the present status of, and development trends in, MOX fuel technology in the areas of design, fabrication, performance, in-core fuel management, transportation, spent MOX fuel management, decommissioning, waste treatment, safeguards and alternative approaches for plutonium recycling is provided. The report concentrates on MOX fuel for thermal power reactors; however, specific aspects of fast reactor MOX fuel are also considered.

The IAEA wishes particularly to thank D. Farrant for his patience and skills in preparing and correcting the many textual contributions and revisions. The IAEA officer responsible for the organization and compilation of this report was V. Onoufriev of the Nuclear Fuel Cycle and Materials Section, Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

EXECUTIVE SUMMARY

This report provides a global overview of the status of and the advances in mixed oxide (MOX) fuel technologies as of December 2000. It also considers the future trends and possibilities for all aspects of the technologies involved.

Strategic considerations (Section 1)

Wherever uranium fuels are used as a source of power, the management, use and disposal of plutonium and plutonium containing fuel and wastes are intrinsic aspects of the technology. Two main approaches limiting access to separated plutonium are possible: reuse of plutonium as MOX fuel in power reactors or storing plutonium materials in some form of secure repository.

In the repository route direct disposal raises serious non-proliferation concerns that will remain for centuries or millennia. This section concentrates on the reactor recycle strategy and, in doing so, identifies three temporal phases in the development of plutonium technology:

Phase 1: The introduction of MOX fuel recycle in light water reactors (LWRs) on an industrial scale.

Phase 2: The situation from now until, say, 2030, characterized by an expansion of MOX recycle to more countries and more plants and leading to a gradual reduction in plutonium stockpiles. The introduction of advanced reactor types (including fast reactors) is a key element of this phase.

Phase 3: A long term and expanded use of MOX fuel, accompanied by greater public acceptance and greater economic attractiveness.

The aim in the third phase will be to manage plutonium stocks effectively. Depending on the state of the nuclear power industry, this could take the form of widespread use of fast reactors or the introduction of incineration plants, respectively either to maintain plutonium stocks or, in the case of a withdrawal from the nuclear era, to remove plutonium and other actinides from the fuel cycle and destroy them.

Plutonium feed production, handling and storage (Section 2)

Numerous methods have been investigated and developed to obtain plutonium dioxide in a form suitable for MOX fuel fabrication. Of these methods, precipitation of plutonium oxalate is the most widely used in commercial manufacturing. This technology is generally considered to be mature.

Of the other processes highlighted, thermal denitration has an attraction from a reduced liquid effluent viewpoint. A variant (microwave heating) is being utilized in Japan. Gel precipitation also has attractions relating to dust minimization and is currently being investigated further in Germany and Switzerland.

The other key area now attracting research investigations is the conversion of plutonium metal to plutonium dioxide, driven largely by weapons disposition programmes. Considerable work in France, the Russian Federation and the United States of America has been, and is still being, devoted to a number of techniques for plutonium metal conversion.

MOX fuel fabrication (Section 3)

Significant MOX fuel fabrication activities have been conducted since the 1950s and commercial scale operations have been pursued since the 1980s. The MOX fuel manufacturing industry has reached maturity as a result of this long operational experience. The large scale MELOX (France), SMP (United Kingdom) and PFPF (Japan) plants are examples of the extension of existing technologies into more advanced and larger facilities. Over the years, fabrication conditions had become more demanding, for example, tighter specifications, more radioactive plutonium to be processed, higher plutonium contents in the MOX fuel to be manufactured and more demanding waste minimization objectives.

The plants could, in general, be used at their full licensed capacity. A total of 1400 t HM of LWR MOX fuel and 150 t HM of fast breeder reactor (FBR) MOX fuel had been produced by December 2000. The lessons learned from this experience have been instrumental not only in designing and starting up the new facilities, but also in backfitting plants that have operated for many years.

In the coming decades, the industry will be confronted by additional challenges, which will again require similar progressive improvements dictated by experience.

LWR fuel assembly design, in-core fuel management and licensing (Section 4)

In-core fuel management with MOX fuel assemblies is common practice in a large number of nuclear power plants in several countries. In six countries, thirty nine commercial reactors with core loadings of typically 30% MOX fuel assemblies are in operation and meet all applicable safety criteria. The experience gained from such cores confirms the suitability of modern design codes for current plutonium concentrations and compositions. The impact of MOX on neutronic design and safety aspects is well understood and this understanding provides the basis for the use of higher plutonium concentrations and, as a consequence, increased discharge burnups. Higher MOX loadings (up to 100%) are being investigated and their feasibility has already been demonstrated.

LWR MOX fuel design and performance (Section 5)

The rod and assembly design of MOX fuel is universally based on, and also follows the evolutions of, uranium fuel design, with only minor modifications at most.

National and collaborative irradiation tests on LWR MOX fuel in normal and off-normal conditions have been carried out over a period of 35 years. These programmes, some of which are still ongoing, are providing the data necessary to compare the behaviour of MOX fuel with that of uranium fuel, and to develop specific MOX fuel performance models. In the same period, over 2000 fuel assemblies have been irradiated in commercial LWRs without any restrictions on plant operating flexibility and fuel reliability. Discharge burnups of up to 50 MW·d/(kg HM) fuel assembly average have currently been reached.

The post-irradiation examinations accompanying both the test programmes and the commercial irradiations have enabled comparison of MOX fuel with uranium fuel. R&D programmes, mainly focused on fission product retention in pellets, are still continuing with the aim of enhancing the fuel performance in terms of discharge burnup without any penalty on operational and safety margins.

Transportation (Section 6)

There has been extensive experience in the transportation of plutonium powder, fuel rods and fresh as well as irradiated MOX fuel assemblies. All have been safely transported for more than 30 years, mainly by road but also by sea and air.

Very rigorous transport safety requirements have been adopted by all countries based on the IAEA Regulations for the Safe Transport of Radioactive Material. These regulations require that the packagings must be safe under both normal and accident conditions of transport and, additionally, that their design must take account of the fissile nature of their contents.

New packagings are being developed in order to take into account the evolution of fuel designs and composition and also to reduce transport and licensing costs.

Spent MOX fuel management (Section 7)

Spent MOX fuel differs from spent uranium fuel principally in having a slower decrease of decay heat generation, a larger fissile inventory and an increased content of minor actinides. All these have to be taken into account for spent nuclear fuel management but they do not prevent the storage and disposition technologies established for uranium fuels being applied to MOX fuels. Both the direct disposal and the reprocessing options can be envisaged as being applicable to spent MOX fuel.

In total, 47 t HM of FBR MOX fuel and 14 t HM of LWR MOX fuel had been reprocessed by December 2000, part of this total carried out in industrial reprocessing plants. Most of the separated plutonium has been refabricated into MOX fuel. This has provided assurance that MOX fuel can be reprocessed in existing reprocessing plants and that the plutonium can be refabricated into MOX fuel with current manufacturing techniques.

Waste treatment and decommissioning (Section 8)

Major considerations in the decommissioning of MOX and plutonium production facilities arise from the presence of plutonium. Plutonium plants have priority for early decommissioning unlike, for example, standard facilities, because they are prone to deterioration of plant and equipment resulting from the high activity present and to an increase of radiation levels as a result of americium in-growth. The principles of risk management are heavily utilized in the process of decommissioning.

A number of countries have already completed decommissioning projects on plutonium processing facilities. There are, however, significant projects to come in the future. These projects will benefit from the experience gained with past facilities and also from early consideration of decommissioning at the design stage. A further advantage of designing for decommissioning is the ability to use remote techniques for standard operations, which may not be possible if consideration is not given prior to construction.

The minimization of plutonium in waste streams is also being planned. Through a process of optimizing waste generating operations, education of personnel as well as requirements from licensing authorities, plutonium levels in wastes have been significantly reduced. Further incentives to reduce waste volumes are also clear, with cost considerations and public acceptability being the drivers in this area.

Application of safeguards and physical protection to MOX fuel (Section 9)

In parallel with the industry improving fabrication plants over the past twenty years, the IAEA and Euratom have continued to develop and improve safeguards, designed and implemented to detect diversion of plutonium for undeclared uses. Such improvements include modern design information handling and analysis techniques, unattended non-destructive assays and remote monitoring systems together with containment and surveillance systems. These improvements have made great contributions to reducing safeguards inspection effort and personnel exposure without losing confidence in their implementation.

Besides international safeguards inspections, physical protection measures are put in place by States to protect plutonium from theft, unauthorized diversion or terrorist attack. A Convention on Physical Protection obliges the Parties to make specific arrangements and meet defined standards, recommended under the auspices of the IAEA. This has been instrumental over the years in preventing the diversion of plutonium from commercial facilities, including reprocessing plants, MOX fabrication plants, nuclear power plants and all systems for the transportation of plutonium bearing materials.

Specific aspects of fast reactor MOX fuel (Section 10)

It is generally agreed that the major mission of fast reactors (breeders and burners) is to provide a long term stable energy supply. One of the key aspects of fast reactor fuel design is the achievement of high burnups. Arising out of the results obtained from experiments with MOX fuel, it is currently believed that the target burnups for future large commercial fast reactors of about 200 GW·d/(t HM) can be achieved if improvements in fuel assembly and rod structural materials are incorporated.

In addition to this, an FBR system could contribute to the reduction of the amount of transuranium materials because of its flexibility in accommodating different fuel types and compositions.

Alternative approaches (Section 11)

One principal planning objective of plutonium management is the provision of protection from unauthorized diversion, while avoiding long term safeguards obligations. Another aspect is allowing the sustainable use of nuclear power and realizing the energy potential of natural uranium. In this context, advanced systems specifically designed for effectively dealing with the production and management of plutonium are under development or investigation. Alternative fuel types (e.g. inert matrix fuel, carbide and nitride fuel) and reactor concepts (FBR and high temperature reactor (HTR)) can also provide for reduced plutonium generation and/or increased plutonium reduction rates. Additionally, alternative systems for transmutation of plutonium and minor actinides into non-fissile waste are under development (accelerator driven systems). The investigation of advanced intrinsically safe systems is a prerequisite for future public acceptance of nuclear power.

To conclude, MOX fuel is currently managed successfully on an industrial scale while its future evolution is under development.

1. STRATEGIC CONSIDERATIONS

1.1. INTRODUCTION

Wherever uranium fuels are used as a source of power, the management, use and disposal of plutonium and plutonium containing fuel and wastes are intrinsic aspects of the technology. The major issues of plutonium technology which are described in this report have acquired an additional dimension from the existence of large amounts of separated weapons grade plutonium or plutonium in the form of weapons components which are to be made inaccessible for further military use.

The two routes, the civil and the military, involving the production, separation and stockpiling of separated plutonium, as well as the management of plutonium contained in reactors or stored as spent fuel, have a common constraint. This is the need to ensure that the plutonium, when not in reactors or when not part of an accepted military programme, is inaccessible for unauthorized and non-peaceful uses. Two main approaches limiting access to separated plutonium are under discussion or in operation. These are, firstly, the reuse of plutonium as mixed oxide (MOX) fuel in power reactors. This approach places the plutonium in a reactor core, then again in a spent fuel environment until it may again be separated and recycled once more through an appropriate reactor system. Secondly there is the concept of putting plutonium materials can-in-canister; highly active glass or synroc blocks or spent fuel to be placed in deep underground repositories in perpetuity. In each case, as the protection provided by the surrounding fission products decreases with time, either for the direct geological disposal strategy or for long term storage of plutonium in spent uranium or MOX fuel, questions arise as to how inaccessible the plutonium will be and for how long the chosen repository will protect it against unlawful access and removal (safeguardability).

One of the major advantages of the reactor spent fuel strategy is that it allows the plutonium to be fully utilized as the energy resource it is, thus adding to the sustainability of the nuclear option. In unlocking this large energy potential, there are three concepts that increase the safeguardability of the plutonium. The first is recycle of the plutonium until it is degraded as a weapons material. This can be realized to a high degree in advanced nuclear plants such as fast reactors [1]. The second is to develop an advanced incineration process that will finally transmute the surplus and exhausted plutonium as it reaches the end of its productive life [2]. The third is to reduce the production of plutonium by avoiding the use of uranium fuels, an option that is in the early stages of development [3].

In choosing the power reactor route for dealing with plutonium management, it is vitally important to develop fuel strategies and fuel cycles which will be efficient in economic power production while at the same time retaining high safety standards and offering a high resistance against unauthorized access to the plutonium.

This issue of the management and disposal of plutonium by its tightly controlled use in power reactors has thus become a significant factor in dealing with the ongoing development of nuclear power. On the one hand, a continued accumulation of surplus plutonium presents a future burden to the world community, not least due to the risk of its diversion for military purposes. On the other hand, only with plutonium may one utilize the practically unlimited energy resources of fertile ^{238}U for which fast reactors will eventually be needed. Plutonium use in thermal reactors also increases the uranium utilization efficiency, but only by a factor of about two. In spite of this, plutonium recycling in light water reactors (LWRs) has evolved to an industrial level and is now a mature technology in several countries [4]. This was due partly to the delays in the development of fast reactors, partly to the resulting need to have an early return on the investment made in large scale reprocessing facilities, partly to the resulting benefit of minimizing high level waste (HLW) volumes requiring geological disposal and, not least, to the need to limit the increasing inventories of separated plutonium.

For the purpose of this report, one can identify three temporal phases in the development of plutonium technology. The first phase has been the introduction of fuel recycle in LWRs accompanied by industrial scale reprocessing as well as design, fabrication and testing of MOX fuel. The second phase, lasting, say, till around 2030, should see the expansion of MOX recycle to more countries and more plants in which the growing stockpiles of plutonium should be reduced. During this period, development of improved fuel cycle facilities and processes for MOX (reprocessing, fuel fabrication, storage and transport), as well as the development of new reactor concepts, must take place in preparation for a third expansion phase. It is important that MOX fuel earns greater public acceptance and demonstrates improved economic performance to justify its expansion if it is to move into the third phase. If these objectives cannot be achieved, there must then be facilities in place by the end of the second phase to allow removal of as much plutonium as possible from all stages of the fuel cycle, by incineration and transmutation.

1.2. PRESENT STATUS

Plutonium has been successfully recycled in the form of MOX fuel in thermal reactors (essentially LWRs) for more than thirty years (Section 4). As of 2001, more than 30 thermal reactors use MOX fuel complying with a partial core loading pattern. Section 3 gives the current status of MOX fuel fabrication worldwide.

The commercial application of MOX fuel in LWRs grew in the mid-1980s as a result of a slowdown, leading later to a virtual abandonment of fast reactor (FBR)[1]

---

[1] Throughout this report FBR refers to fast reactor (breeder or burner).

programmes in many countries. As a result, the technologies of recycling and fuel fabrication were then adapted for plutonium recycling as LWR fuel, focusing, in so doing, on stabilization of the separated plutonium inventory.

Currently, the use of MOX fuel has been established on an industrial scale in a number of countries. In Belgium, France, Germany, Japan and Switzerland, a considerable number of thermal power reactors, both pressurized water reactors (PWRs) and boiling water reactors (BWRs), are either licensed, or a license has been applied for, to use MOX fuel at levels of up to 30% or more of the reactor core (Section 4).

Reprocessing capacity today amounts to some 3400 t HM/a in the Organisation for Economic Co-operation and Development (OECD) countries, essentially commercial, and some 620 t HM/a in non-OECD countries (Russian Federation and India), especially at non-commercial plants (Chelyabinsk RT-1, Russian Federation; Tarapur PREFRE-1 and Trombay, India). It is notable that China will join the reprocessing community, while India, Japan and the Russian Federation may expand their reprocessing capacities by commissioning new facilities in the coming decade. Meanwhile, the Sellafield B205 Magnox plant in the United Kingdom (UK), with a capacity of 1500 t HM/a, is scheduled to be shut down around 2012. Forecasts of worldwide reprocessing capacity after 2015 amount to some 6000 t HM/a. This reprocessing capacity is sufficient according to current projections of requirements.

About 10 500 t HM of spent fuel were discharged from nuclear power reactors and about 2900 t HM of spent fuel were reprocessed in 2000, which corresponds to about 28% of the total. About 20 t of plutonium were separated in reprocessing plants and 9 t were used mainly as MOX fuel in LWRs. The imbalance between the separation and the use of plutonium as MOX fuel had resulted in an accumulated inventory of separated civil plutonium of about 180 t at the end of 1998, increasing to about 210 t at the end of 1999 and about 220 t at the end of 2000. These data evaluations are based on the declarations of IAEA Member States [5].

Utilization of MOX fuel only in thermal power reactors will not ultimately resolve the issues related to accumulation of quantities of discharged spent fuel and separated plutonium requiring to be stored. In recent years, not more than 30% of the fuel discharged from thermal reactors has been reprocessed while some 40% of separated plutonium has been used in MOX fuel fabrication. This is due to the limited fabrication capacities and lack of facilities for in-house recycle or to restrictions on the use of plutonium from other sources. The five plants currently fabricating MOX fuel are in Belgium, France, Japan and India (Section 3) with a total capacity amounting to some 210 t HM/a. A sixth plant is being commissioned in the UK. Additional MOX plants are being planned in Japan and China.

More efficient use of plutonium will ultimately be made in fast reactors, where multiple recycling is possible and has been demonstrated. In spite of the slowing down of fast reactor development, the long term need for this type of reactor has been recognized. In the Russian Federation, it is intended eventually to recycle plutonium in commercial FBRs and there are plans to construct three such reactors [6].

1.3. MEDIUM AND LONG TERM TRENDS

Three temporal phases in the management of plutonium arising from nuclear power generation (civil plutonium) were identified above.

The first phase, outlined in Section 1.2, which has seen a gradual introduction of the recycling of mixed oxide fuel up to an industrial scale in light water reactors in several countries, is currently well established. MOX usage in LWRs is a well understood and mature technology and MOX can be managed in a very similar way to uranium fuel. The current inventory of separated civil plutonium is safely stored (Section 2) and has still to be recycled. The necessary safeguards arrangements have kept pace with these developments wherever plutonium is handled, fabricated, stored, transported and irradiated.

The second phase (up to, say, 2030) is beginning at the time of writing. It is characterized by an ongoing but irregular expansion of MOX recycling in which other countries will acquire MOX technology and additional nuclear plants will be licensed for MOX fuel. It could then be expected that the stockpiles of separated plutonium (both civil and weapons surplus) will begin to diminish and move to significantly lower levels during this phase. As well as extending the number of power plants loading MOX fuel, the technology now needs to be developed for advanced reactors such as the fast reactor advanced BWRs (ABWRs) with 100% MOX cores and the burning of weapons grade plutonium. Additionally, more environmentally friendly reprocessing methods with reduced emissions will be needed to prepare for the third and long term phase of plutonium recycle and management.

In this preparation for the third phase of development, emphasis must be placed on both achieving greater public acceptance for plutonium fuel technology as well as making the power producing utilities more attractive economically. These same utilities will be the ultimate source of the financing needed to establish the stable development of nuclear power based on uranium and, over the longer term, plutonium fuels. A fuel that is more expensive than others or which may limit reactor operation is unacceptable for the power producers. If the economic and performance targets cannot be reached and negative public attitudes persist, there will be no third phase of plutonium recycle.

However, if such a development can be realized, the third phase should see the progressive introduction of advanced thermal reactors as well as fast reactors, alongside new technologies specifically designed for MOX fuel. The aim will be to manage the production and use of plutonium effectively to avoid major stockpiling. Depending on whether nuclear power is generally increasing, stable or decreasing, the introduction of breeder reactors or plants designed to ‘incinerate’ the surplus plutonium will be selected according to need. Overall, the aim will also be to keep final waste repositories ‘safeguards-free’ by the exclusion of plutonium from them. All of this will require an efficient, safe and economic reprocessing industry coupled with the necessary international safeguards regime [7].

If nothing is done in this direction, under current arrangements all residual spent fuel, including MOX, must be disposed of by medium term interim storage followed some decades later by deposition in a final repository. This is simply because at the end of its life, some six to eight years prior to plant closure, no more recycling in the parent plant is possible. Even the transfer to another, still operating, plant only delays the inevitable final disposal of fuel assemblies.

Returning to the opening discussion, for MOX fuel the question arises as to whether the direct disposal route described here has been sufficiently thought through regarding the non-proliferation and safeguards aspects. The situation is even worse for spent uranium fuel, since here the plutonium quality makes it more desirable for unauthorized use. Owing to the decrease of radioactivity, the residual fissile material (plutonium) consequently becomes more accessible with time and safeguards measures (accounting, containment and surveillance) need to be applied indefinitely, in particular where spent fuel is finally stored. This is even more the case if such repositories are of the retrievable type. Can such a long term surveillance and control be guaranteed over centuries or millennia [8] and are governments or societies sufficiently stable to adhere to their non-proliferation commitments over centuries?

1.4. NUCLEAR FUEL CYCLE BACK END CHOICES

If direct disposal places an unacceptable and practically impossible open ended commitment on society to monitor and guard spent fuel repositories in perpetuity (unlike those for radioactive wastes and residues), then an alternative must be sought by separating plutonium from spent fuel and returning it as quickly as possible into a reactor where it is truly inaccessible. There are two scenarios to consider:

(1) Declining nuclear power industry, either regionally or worldwide

- The need is for the development and deployment of advanced fuels and reactor systems for the complete, or nearly complete, destruction of plutonium and actinides in so called incineration plants. Some decades can be allowed (interim spent fuel storage) until these systems and materials have been fully developed.
- For those countries abandoning nuclear power, an international solution (e.g. fuel cycle centres) will have to be found.

- In this scenario there continues to be a strong need for reprocessing of spent fuel. For non-proliferation reasons alone, reprocessing cannot be abandoned.

(2) Expanding nuclear power industry

- Efficient plutonium fuelled reactors are required which can match an expanding fuel demand by breeding, extraction and recycle of unused fertile/fissile material. At the same time, the strong non-proliferation characteristics of plutonium ‘storage’ in a power producing plant can be achieved. FBRs and advanced converter reactors will be the mainstay of this scenario.
- In view of the need for a viable and expanding reprocessing industry to support this concept, advanced techniques must be developed (as in the preceding case) to allow low cost and low emission ‘fissile extraction’ processes and advanced fuel fabrication facilities to be developed, all with low waste arisings.
- A strong research and development component is also implied, but this can only be realized if the sources of financing are assured. The industry relies for its research and development financing directly or indirectly on its utility customers. If the utility customers cannot survive in liberalized power markets, the postulated expansions will not be sustained and the source of industry funding needed for the advanced reactor and recycle technologies will disappear.

It must be left to governments and society as a whole to consider their obligations and the extent they may be required to support (subsidize) either the development of advanced systems, simultaneously putting the necessary agreements and controls in place to secure a robust proliferation-resistant fuel cycle regime, in support of nuclear power, or the means for ensuring a controlled and safe abandonment of nuclear power.

The remainder of this report will look at the mature state of technology today and its evolutionary development to support the second phase of plutonium recycle with efficient and economic processes and facilities to bring the stockpiles into balance at a much lower level. The report concludes with a review of the revolutionary developments needed if there is to be a long term and expanding era of nuclear power fuelled by plutonium. But the future demands also that equal attention be paid to the development of the technology to destroy plutonium if the nuclear industry were to be shrinking. Geological disposal of plutonium is not a long term option!

2. PLUTONIUM FEED PRODUCTION, HANDLING AND STORAGE

2.1. INTRODUCTION

The fabrication of MOX fuel requires plutonium in oxide form and therefore the issues of its production, handling and storage in a safe and secure manner, under international safeguards, are important. The contents of this section cover the present status, as well as current and future issues, of plutonium feed production, handling and storage. The methods of plutonium dioxide production described herein are limited to the final conversion step to plutonium dioxide; the separation processes themselves are discussed in a similar technical document associated with reprocessing.

There are a number of different methods that can be used to produce plutonium dioxide, including the methods used for uranium oxide production. They may start from either plutonium nitrate, derived from the dissolution of irradiated fuel in nitric acid, or from plutonium metal, recovered from weapons material, for instance. The methods listed below are discussed in greater detail in Section 2.2.

(a) Methods of converting plutonium nitrate to plutonium dioxide include:

- precipitation methods (precipitation of plutonium oxalate)
- thermal de-nitration methods
- co-precipitation methods
- gel precipitation methods.

(b) Methods of converting plutonium metal to plutonium dioxide include:

- oxidation
- conversion to an intermediate species prior to conversion to dioxide.

There are significant issues regarding the handling and storage of plutonium dioxide. The storage of plutonium is, in general, a sensitive and political issue. Storage clearly has different aims dependent on the length of time and purpose of the storage. The important issue of very long term safeguards measures, which provide confidence in the storage methods, was raised in the introductory section. This section deals only with the safe storage of plutonium dioxide powder prior to fuel fabrication and does not cover the storage of plutonium, either as a metal [9] or in a final form, via vitrification [10] or encapsulation.

2.2. PLUTONIUM FEED PRODUCTION

Numerous methods have been used, and others are being developed, to obtain plutonium dioxide in a suitable form for MOX fuel fabrication. The methods, deriving plutonium dioxide from the nitrate, are considered first, followed by the processes to convert plutonium metal to oxide powder. The status of each process, commercial or developmental, is indicated.

2.2.1. Precipitation of plutonium oxalate

This precipitation method is the most widely used commercial method of converting plutonium nitrate solution to dioxide and is summarized [11] in Fig. 1. Precipitation techniques superseded the de-nitration process (see below) as they offered further purification as well as better handling properties of the final product. This was particularly important when the driving force for plutonium finishing was military, as the performance of weapons grade material depends on the isotopic and chemical purities.

The process is described by Eqs (1) and (2). The plutonium stream is concentrated in evaporators and conditioned to Pu (IV) with hydrogen peroxide before precipitation. An excess of oxalic acid is used to reduce the solubility of the product. The oxalate is decomposed to oxide, which is suitable for ceramic fuel fabrication. The particle size and structure of the oxide are governed by the initial

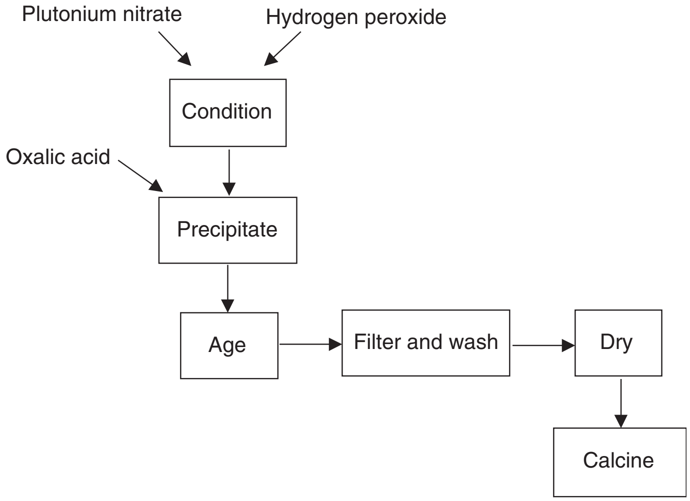

FIG. 1. Flow sheet of production of plutonium dioxide via oxalate precipitation.

precipitation and, in turn, by the temperature and by concentrations of plutonium, nitric acid and oxalic acid,

$$Pu(NO_3)_4\cdot 6H_2O + 3H_2C_2O_4 \rightarrow Pu(C_2O_4)_2\cdot 6H_2O + 4HNO_3 \tag{1}$$

$$Pu(C_2O_4)_2\cdot 6H_2O \rightarrow PuO_2 2CO + 2CO_2 + 6H_2O \tag{2}$$

The process can be optimized to give plutonium dioxide properties suitable for the requirements of the end product. In the region of calcination temperatures of about 600°C, small variations in temperature produce large differences in specific surface area (SSA) (Fig. 2). This parameter affects the pressing characteristics of the feed and also affects storage volumes and conditions. For example, powder with high SSA values can adsorb increased quantities of moisture and gas prior to canning. As the powder heats up on storage, any adsorbed moisture could lead to pressurization of the can. The current conditioning methods used in the United States of America (USA) produce plutonium dioxide unsuitable for MOX production but in the UK less stringent calcining conditions are used to achieve a more suitable feed powder while still achieving an oxide with satisfactorily low water content. In France there is less of a requirement to consider the long term storage of plutonium dioxide as there it is utilized in MOX fuel fabrication soon after production.

Chemical impurities can be an issue in powder processing since commercial reactor fuel specifications with respect to composition are very stringent. Impurities can also affect the processing behaviour of the feed powder. Residual carbon in particular, which arises from incomplete conversion of oxalate to oxide, pick-up or

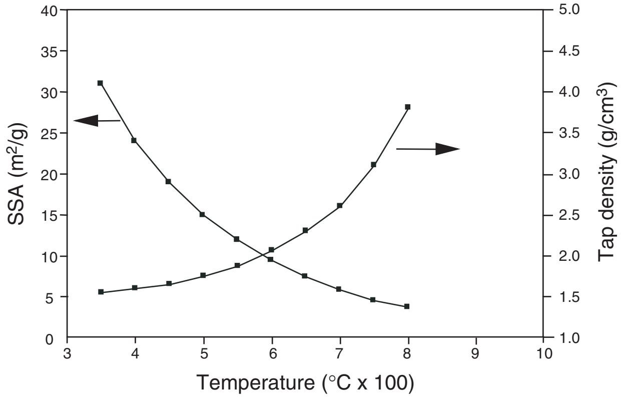

FIG. 2. Effect of temperature on specific surface area and tap density of plutonium dioxide powders.

inefficient removal during heat treatment, can inhibit sintering and give rise to solarization in pellets (i.e. fuel swelling rather than densification). In this process, the decontamination factors are significant for most impurities.

2.2.2. Thermal de-nitration methods

As with uranium, the most straightforward way of converting from a nitrate to an oxide is by direct thermal degradation (Eq. (3)):

$$Pu(NO_3)_4 \cdot 6H_2O \rightarrow PuO_2 + 4NO_2 + O_2 + 6H_2O \qquad (3)$$

However, such methods are not presently used on a large scale in European plutonium feed production plants although wider use of them is being made in Japanese plants.

Thermal de-nitration (TDN) of the oxalate has potentially the minimum number of stages and produces a low volume of liquid effluents that require further treatment. However, such benefits are outweighed by the micron particle powder quality of plutonium dioxide produced by direct TDN, rather than the free flowing powder generally desirable for subsequent processing.

Different countries have investigated different de-nitration processes. The French NITROX process carries out the dehydration under reduced pressure to keep the temperature below the melting point of the hydrates. In the USA, the use of ammonium nitrate to form a mixed salt to modify the decomposition path has been patented as an alternative process. In Japan the commercial use of microwaves has been patented in the drying and de-nitration stages [12], and this has been applied directly to produce a (U–50%Pu)O_2 raw powder. The decision to produce co-denitrated uranium/plutonium oxides stems from the agreement between Japan and the USA to reduce the potential risk of proliferation. The use of plasma methods has been reported in the Russian Federation [11]. In the UK the use of cover gases, such as NO_2 and CO_2, to displace water from uranyl nitrate hexahydrate has been investigated and the use of a similar process has been proposed for plutonium. This has been used on a laboratory scale [11] but is not suitable for commercial use.

2.2.3. Co-precipitation methods

A solid solution of MOX fuel would be better produced by starting from mixed aqueous solutions rather than by dry blending oxides and relying on solid state diffusion during sintering, hence co-precipitation methods were considered. MOX pellets from co-precipitated feeds form a solid solution at about 1000°C in hydrogen and are completely soluble in nitric acid up to 40% Pu/HM. Pellets prepared from mechanically mixed powders form solid solutions at about 1600°C, but there are recurrent problems with the specifications formulated for solubility in nitric acid.

Co-precipitation of both uranium and plutonium, as ammonium diuranate (ADU) and plutonium hydroxide, from mixed nitrates with ammonia gas in a nitrogen carrier gas is a process similar to the production of UO_2 feed powder via the ADU process, which is still a commercial finishing step. Unfortunately, plutonium and uranium precipitate at different pH values. In the UK a multi-stage co-precipitation route with ammonia in a pH gradient was adopted, but this still did not give an intimately mixed product and so the process was discontinued [11]. Further developments in the Russian Federation, using co-precipitation techniques with ammonium hydroxide and polyacrylamide ('GRANAT'), are reported in Ref. [13].

One US method to finish with a combined uranium and plutonium stream was the so-called CO-PREcipitation and CALcination (COPRECAL) route [14]. Slurry from ammonium co-precipitation was fed directly to a fluidized bed for thermal decomposition. Again this process was never implemented commercially, perhaps because of the development of gel sphere co-precipitation routes (Section 2.2.4).

Workers in Germany [15] extended the ammonium uranyl carbonate (AUC) precipitation route to form the ammonium uranyl–plutonyl carbonate (AUPuC) process, using carbon dioxide to precipitate isomorphous mixed compounds of uranium and plutonium, provided both are hexavalent. Uranyl nitrate was mixed with 40% plutonium nitrate solution for direct production of a master blend. Large soft particles could be diluted to the appropriate concentration with free flowing AUC derived UO_2 and directly pelletized, avoiding dusty and dose intensive milling and conditioning stages. The process was used industrially in the Hanau plant to convert the plutonium nitrate shipped from the WAK (Wiederaufarbeitungsanlage Karlsruhe) reprocessing plant and to chemically recover the MOX fabrication scraps. However, owing to the withdrawal of the operating licence, the plant has since been closed and the process abandoned.

2.2.4. Gel precipitation methods

Gel supported precipitation followed co-precipitation methods and while this has been, for the most part, only investigated on an experimental scale, some commercial production has taken place. The methods, varying from country to country, essentially involved drops of a mixed nitrate solution with an organic polymer, such as polyacrylamide, and a structure modifying agent, such as formamide. A precipitate is formed by hydrolysis. The internal and external gelation routes are defined by the formation of the hydrolysing base, either internally or externally. The spheres are aged, washed and dried in further columns before calcination and sintering. The main advantage, because the process is carried out in liquids, is the avoidance of dust formation and the consequent radiation problems. However, the method produces significantly greater active liquid effluent [11].

The spheres produced by these methods allow an alternative method of rod filling, i.e. being vibro-packed into the fuel rod, rather than pressed into pellets. Using different sizes of spheres, which can be sintered to around 95% of the theoretical density, smear densities of around 80% can be achieved in pins. In the UK, gel precipitation methods were developed to fabricate MOX fuel for the Prototype Fast Reactor (PFR) at Dounreay. Initially the gel spheres were used as a direct feed using vibro-packing techniques to fill the pins. However, it was found that the different distribution of void space led to anomalous corrosion and fuel slumping at the very high ratings in fast reactors. A route using the spheres as a press feed for the manufacture of annular pellets was later adopted as a consequence.

Work on gel precipitation methods for MOX fuel fabrication continues today. The Paul Scherrer Institute (PSI) in Switzerland uses an internal gelation route and the Institute of Transuranic Elements (ITU) in Germany is developing sol–gel processes [16, 17].

2.2.5. Methods of conversion of plutonium metal to oxide

The conversion of plutonium metal to oxide powder is a medium term issue linked to the recovery of metal from the disarmament process, but it is worth mentioning here. As more weapons plutonium becomes available for use in a civilian context, the more interest there is in converting it to an oxide form suitable for fuel fabrication. The conversion may take place via direct oxidation or oxidation after conversion to an intermediate product, through any of the processes mentioned previously in Section 2.2.

Plutonium metal will oxidize in an uncontrolled manner if exposed to an oxidizing atmosphere. The general method of conversion is oxidation of the metal to plutonium dioxide, via roasting or calcining [10]. Plutonium metal within waste streams, such as casting skulls and turnings, can be converted to an oxide form by first burning under controlled conditions, then dissolving the substance in HNO_3–HF to produce plutonium nitrate. This can then be converted to an oxide by one of the methods previously discussed in Section 2.2 [18].

There are technical issues regarding the conversion of weapons grade plutonium into MOX fuel. One issue arises from the presence of metallic gallium, used as an alloying addition in weapons grade plutonium (WPu) [19, 20]. Although most of the gallium volatilizes during the sintering process, that which remains reacts with many metals. The effect of gallium on MOX fuel production, in terms of its effects on fuel and cladding materials, is therefore important. Information on these effects leads to the determination of an optimum route of plutonium separation from gallium.

Development work is currently being performed in the Russian Federation in the area of conversion of metallic plutonium to plutonium dioxide. A molten salt method [21] is being developed in particular, although so far only successfully for vibro-packed fuel. Work to overcome problems in producing pelletized fuel from this method continues. The advantages and disadvantages of other methods for the conversion of plutonium metal to oxide such as aqueous chemical techniques (oxalate and ammonium precipitation and co-precipitation) and other pyrochemical techniques are also discussed in Ref. [21]. A method of reprocessing mixed U–Pu fuel is being developed in the Russian Federation consisting of dissolution of the fuel in molten molybdates and precipitation of homogeneous solid solutions of $(U–Pu)O_2$ [22].

Such research and development studies aim to assess a suitable conversion process in terms of performance, safety and secondary waste management [23].

2.2.6. Feed production finishing lines

The finishing lines of a reprocessing plant produce a final product of plutonium dioxide powder. Batches are typically 90–100 kg plutonium. The plutonium dioxide powder is then placed into cans; La Hague cans contain 2.7–3.2 kg plutonium and Sellafield cans 5–7 kg plutonium [24]. The Japan Nuclear Cycle Development Institute (JNC) utilizes a two level layer type container for plutonium storage both in product storage at its Tokai Reprocessing Plant (TRP) and feed storage at the Plutonium Fuel Production Facility (PFPF). The inner can has a specially manufactured screw top configuration providing a leak tight capability. This can accommodates approximately 2.5 kg MOX powder with Pu/HM content of 50%. The outer container is a canister that features a double door lid system for the interface with the head end of the process glovebox. One canister can hold four inner cans totalling approximately 10 kg MOX or 5 kg PuO_2.

The isotopic composition of the final product from the finishing line clearly depends on the derived source of plutonium. Within a large reprocessing plant it is impractical to supply customers with plutonium specifically from their own spent fuel assemblies. In reality, the sources of plutonium are mixed and customers receive an amount of plutonium related to the fissile content of the spent fuel they are having reprocessed. Precipitation and finishing of plutonium dioxide is a batch operation at La Hague, ensuring identical isotopic composition through the batch, but it is a continuous process at Sellafield. Blending is required to ensure the contents of plutonium cans have a uniform isotopic composition. The homogeneity of the plutonium isotopic composition is important in terms of MOX fuel fabrication for nuclear design and core management considerations. This is discussed in Section 3.

2.3. HANDLING AND STORAGE OF PLUTONIUM DIOXIDE POWDER

Criteria for storage have been developed based on the potential hazards of the substance. These criteria do not dictate specific design features and different organizations use different methods for achieving the same objective. The issues covered by the criteria are packaging, containment, structural and pressure requirements, and inspection [9].

Owing to the radiotoxicity of plutonium, plutonium dioxide is conveyed in sealed vessels or handled in gloveboxes at subatmospheric pressures. Remote handling of plutonium is employed whenever possible, particularly in large reprocessing plants, both before and after it has been loaded into cans. Stringent plutonium accountancy for safeguards and safety reasons imply the handling of plutonium is more expensive than that of uranium.

2.3.1. Evolution of storage methodology

There has been a significant evolution of the aims of storage over time, on the realization that long term storage was required and that the proposed short term storage plans could not be achieved. Since then, stores have evolved in terms of greater safety and more safeguards systems rather than changes in approach. Recent stores consist of large thick-walled concrete cells containing re-entrant tubes accessible only through loading ports. The methodologies adopted by different countries basically fall into two types: the horizontal channels utilized in the UK in the Thermal Oxide Reprocessing Plant (THORP) and the vertical channels favoured by the USA and France as used at La Hague. It should be pointed out, though, that the French use this store as a short term measure only. Dependent on the fuel content, decay heat requires the store to be cooled, by forced draught for instance. The stores are seismically qualified and routinely inspected.

2.3.2. Potential hazards associated with storage and handling

The potential hazards associated with storage of plutonium dioxide powder are radiological considerations, criticality, temperature rises and release of airborne contamination due to leakage. The store must also be able to withstand specified potential external events. The awareness of such hazards is documented in literature such as Ref. [25].

The radiological considerations of the feed powder arise from the in-growth of ^{241}Am and its gamma emissions, the alpha emissions from the plutonium isotopes and ^{241}Am and the fast neutrons due to prompt fission. These are countered by shielding and limited exposure of personnel to the packages, i.e. remote handling. The effect of ionizing radiation on the packaging materials has been extensively studied. Problems have been encountered in early experience of storing low burnup Magnox plutonium dioxide in the UK, owing to the deterioration of polythene intermediate bags [7].

Criticality is controlled by the fixed geometry of the store. This was an issue in early stores that consisted of racks, where collapse of the structure could lead to loss of the required geometry. Handling equipment is designed to accommodate only one container, ensuring the masses involved are always subcritical. In addition, any temperature rises are countered by forced air cooling systems. Containment of the package is designed to prevent leakage, with filter systems protecting the outside environment against any airborne contamination that could result from accidents. Clearly the design of the structure of a store takes into account accident and unusual conditions. Additionally, there are laboratory scale development activities in the Russian Federation investigating the technology for fabricating criticality-safe ceramic compositions of plutonium dioxide with different neutron poison materials (such as hafnium, gadolinium, lithium and boron) to ensure subcritical conditions during storage [26].

The major concern over the storage cans used is the potential for pressurization of the can, which is usually the net result of radiolysis of water, though there are contributions from the release of helium generated in the fuel, by radioactive decay of the plutonium isotopes and ^{241}Am.

The presence of ^{241}Am raises issues for the length of storage of plutonium oxide powder before MOX fuel fabrication. Forming from the radioactive beta decay of ^{241}Pu, it is a gamma emitter presenting problems for shielding and dose uptake considerations in fabrication plants. ^{241}Am is also a neutron absorber and reduces the fissile potential of the oxide powder. The isotopic composition of the plutonium dioxide is therefore a limiting factor in the time periods involved for long term storage. Plutonium from different sources and burnups, such as LWRs, weapons or low burnup UK Magnox reactors, contains varying amounts of ^{241}Pu and therefore the time they may be stored before some form of purification is required prior to MOX fabrication varies.

2.3.3. Description of storage containers

Long term storage may be required before the fabrication into MOX fuel.

The procedure for storage of plutonium from reprocessing in THORP at Sellafield in the UK involves cans with three levels of containment. The cans are added to the store by remote handling.

The inner and outer layers are always metallic and the intermediate layer may be plastic (with the attendant inconveniences mentioned in Section 2.3.2) or metallic. Three layers of close fitting steel also provide good thermal conductivity of the packages as well as strength. The primary containment is a stainless steel screw-top can, which holds 5–7 kg of powder. Oxide from the finishing lines is remotely metered into storage cans under an inert atmosphere via a blender. The inner can, with a screwed lid, is accountancy weighed and welded into an intermediate can using a carbon dioxide laser. This intermediate can is then welded inside an outer can. Each can is tested and monitored for external contamination prior to being transferred to the buffer store either for storage or export. Cans used in the UK have a design life of at least 50 years. Examples of these cans are shown in Fig. 3. It should be noted that the inner container is not considered gas-tight, but provides a powder-tight seal. This is a safety consideration in terms of pressurization as the powder can still be contained, even if the outer container were to pressurize and fail.

If heading for storage, packages are taken from the buffer store and conveyed through the package transfer area, which classifies the product prior to it entering the store, via verification, can weighing and can monitoring stations. If packages are to be exported, they also pass through package transfer and the same verification before being leak tested and monitored for external contamination. They are loaded into export flasks prior to transportation from the export area [27].

The same basic principles for plutonium packaging are used in France. The plutonium dioxide powder is put in a stainless steel can, which is then sealed by crimping, and complete containment is achieved through two successive stainless steel containers with welded lids [7].

FIG. 3. Storage cans for plutonium dioxide powder (THORP).

At the JNC reprocessing finishing line (TRP), MOX powder is blended to obtain a homogeneous product and fed into an inner can. After accountancy measurement and decontamination, it is accommodated in a canister. The canisters are conveyed to product storage by remote operation where storage pits, designed for criticality safety, are arranged on the floor.

When a canister is transferred to the MOX fabrication plant (PFPF), a specially designed transportation cask is utilized. After unpacking at the loading area in PFPF, it is remotely transferred to a feed store, which is of a similar construction to that of the reprocessing product store. Dose is minimized throughout by ensuring all transfers occur by remote control techniques and that safeguards accountancy is carried out in an 'inspector unattended mode'. A picture of the storage hall and the fully automated transfer system is shown in Fig. 4.

There are many similarities between the JNC storage capabilities and those being constructed at Japan Nuclear Fuels Ltd (JNFL) sites. Product storage at the Rokkasho Reprocessing Plant (RRP) will utilize almost the same type of cans and

FIG. 4. JNC storage hall and the fully automated transfer system.

canisters as those used by JNC. The design of the storage area and the operating procedure are also similar to those at JNC. At the JNFL MOX (J-MOX) plant (Section 3) to be located at the Rokkasho site it is planned to have similar feed storage, limited, however, to the buffer store scale.

2.4. LONGER TERM DEVELOPMENTS

Development continues on the techniques of plutonium feed production. A process minimizing waste, yet producing feed powder of high quality for MOX fuel production, is the aim. Work in this area includes the gel precipitation work at ITU [16], which produces homogeneous MOX fuel.

Work is also planned on the development of methods for conversion of plutonium metal to plutonium dioxide for use in civilian reactors or storage scenarios, including the Russian work on molten salt processes [21, 22]. The 130 t MOX/a J-MOX plant, on which construction is due to begin in 2004 (operational in 2009) at the Rokkasho site, means that the Japanese will have an increasing interest in the areas of plutonium production and storage.

2.5. CONCLUSIONS

Precipitation of plutonium oxalate is the most widely used commercial process for obtaining plutonium dioxide for MOX fuel fabrication, and this technology has reached an advanced level of maturity with few outstanding issues. Impurity levels and specific surface area are two characteristics where improvements could bring further benefits. Incentives for powders with reduced impurity levels and adequate specific surface area (for minimization of storage limitations) are clear.

Of the other processes highlighted, while each clearly has its benefits and problems (e.g. thermal de-nitration gives reduced liquid effluent but there are concerns over powder quality; gel precipitation gives reduced dust levels but increased liquid effluents), the innovative microwave heating process utilized in Japan offers good potential for the future.

Investigations into the conversion of plutonium metal to plutonium dioxide (in support of the weapons disposition programmes) are now attracting considerable interest. The issues arising from the presence of gallium in metallic plutonium are guiding development programmes to achieve an optimum separation route.

Finally, with regard to plutonium dioxide powder storage, while this is also considered to be a mature technology, interest in storage reduction in general will continue to be a guiding principle with possible minimization of can pressurization (from moisture uptake) being a specific topic of future exploration.

3. MOX FUEL FABRICATION

Overviews of the historical evolution of MOX fuel fabrication and the resulting achievements have been adequately covered in an OECD NEA publication [7], in a presentation at TOPFUEL'97 [28] and at an International Symposium on MOX Fuel Cycle Technologies [29]. These are complemented by several, more limited scope, presentations at various conferences, see for example Refs [30–37]. Relevant parts of these publications are summarized here. (The important safeguards aspects of fuel fabrication are discussed in Section 9.)

This section deals with the fabrication of MOX fuels for LWRs (including the closely related ATR fuel) as well as for FBRs. It is recognized that the two types of fuel have quite different characteristics, which have an impact on both the fabrication processes and the quality requirements [29]. Most industrial fabrication plants are therefore devoted exclusively to either LWR or FBR fuel. Dual purpose facilities have, nevertheless, operated successfully in the past (Belgonucleaire/Dessel, Siemens/Hanau and Cogema/Cadarache) and are still under consideration for future plants (Russian Federation).

3.1. PRESENT STATUS

3.1.1. Fabrication capacities

Significant MOX fuel fabrication activities, in advance of industrial deployments, have been conducted since the 1950s in Belgium and the USA, since the 1960s in France, Germany, Japan, the Russian Federation and the UK, and since the 1970s in India.

In the USA, five pilot facilities, with a combined fabrication capacity of 50–70 t HM/a, were in operation up to 1976 when President Carter took the political decision to defer reprocessing indefinitely. As a result, the facilities were shut down and decommissioned. Most of this valuable experience is now considered out of date.

In Germany, the Siemens/Hanau plant, which started operation in 1972 as a dual purpose (FBR and LWR) facility, reached an effective capacity of 20–25 t HM/a of LWR fuel in the 1987–1991 period. It was shut down by order of the Hesse Ministry of Environment at the time of a contamination incident on 19 June 1991. This plant is now being decommissioned. Since the operating experience is recent enough still to have some relevance today, further mention of this plant is included here. A larger plant (120 t HM/a), constructed on the same site and which was almost ready to start operation, never received a licence to operate and is now abandoned.

The Belgonucleaire plant started operation in 1973 on the basis of R&D conducted over the previous fourteen years in successive facilities from laboratory to pilot scale and also pursued during the first few years of operation of the plant. During the initial period of operation (10 years), the plant was equipped to fabricate both FBR and LWR fuel and did indeed fabricate both types of fuel. On the basis of lessons learned during that period, the plant was temporarily shut down for refurbishment in 1984 and the capacity upgraded to 35 t HM/a for LWR fuel, the only fuel to be manufactured thereafter. Since the mid-1990s, the plant has been backfitted, without interrupting fabrication, to incorporate improvements resulting from accumulated experience and deemed necessary to meet more challenging future requirements.

During the first ten years of operation, the end product of the plant was LWR and FBR fuel assemblies (FAs). Since the mid-1980s, the end product has been fuel rods (FRs), with assembly being performed in the adjacent Franco–Belge de fabrication de combustible (FBFC) international (Dessel) uranium fuel manufacturing plant, to benefit from their large scale production capacity (800–900 FAs/a). The FBFC international plant is equipped with a dedicated MOX fuel assembly, control and storage facility devoted to the LWR FRs produced by Belgonucleaire and Centre de fabrication de Cadarache (CFCa). Their experience dates back to 1963, with the manufacture of the first PWR MOX FA incorporating FRs fabricated by Belgonucleaire.

The CFCa plant, now renamed Cogema/Cadarache, the ultimate achievement of a facility that started operation on a pilot scale in 1962, is devoted to FBR fuels. In the mid-1980s, with Electricité de France's (EDF) decision to utilize MOX fuel in their PWRs, one of the FBR lines was converted to LWR fuel fabrication and started operation on PWR fuel in 1990. More recently, the second fabrication line, up to then still devoted to FBR fuel, has been modified into a dual purpose facility, capable of fabricating either FBR or LWR fuel. The resulting capacity is 40 t HM/a for LWR fuel if no FBR fuel is being fabricated. The manufacturing is performed within CFCa up to finished FAs for FBR fuel but only up to FRs for LWR fuel, the assembly being performed at FBFC International or the Melange Oxides (MELOX) fabrication plant at Marcoule.

A decision to construct the MELOX/Marcoule plant was made in 1985, as a consequence of EDF's decision to load MOX fuel in their 900 MWe PWRs. The plant was originally planned for this purpose only. MOX operation started in 1995 and took over the fabrications for EDF that had been conducted at Belgonucleaire and CFCa during the interim period of construction of MELOX. From this perspective, the plant was licensed for a production of 100 t HM/a. MELOX benefits from a very high level of automation allowing large scale fabrication while minimizing personnel dose uptake. As planned, Cogema recently fitted out the MELOX plant for the fabrication of both BWR and foreign PWR fuel and enhanced the capacity to potentially 250 t HM/a. Details of these developments are provided in Refs [32, 35].

The UKAEA produced 13 t HM FBR fuel between 1970 and 1988 in a now decommissioned plant at Sellafield. In the early 1990s, UKAEA, acting for BNFL, equipped the MOX Demonstration Facility (MDF) in a building at Sellafield that housed MOX fuel development facilities. Both MDF and the UKAEA team were soon thereafter incorporated into BNFL. The purpose of MDF was to gain commercial experience on manufacturing and inspecting thermal MOX fuel in preparation for the large scale Sellafield MOX Plant (SMP) project. It operated in this framework until 1999 when it was shut down because of quality related issues. In the future, when MDF reopens, it will be used as a support facility to SMP, an automated 120 t HM/a book capacity plant described further in Section 3.3. BNFL has proceeded since 1999 with uranium commissioning of SMP but, at the December 2000 status date of this report, was still awaiting the required licence to introduce plutonium into the facility.

In India, a pilot scale fabrication plant, Bhabha Atomic Research Center (BARC), is in operation at Tarapur, producing only BWR fuel at present, but foreseen to be adapted later to produce CANDU and FBR fuel [38].

After conducting development for some years on a laboratory scale in the Plutonium Fuel Development Facility (PFDF) (which is still in operation), JNC started the Plutonium Fuel Fabrication Facility (PFFF) with two completely separate lines — one for Advanced Thermal Reactor (ATR) fuel (1972), still in operation, and one for FBR fuel (1973), which was shut down in 1987. The latter was replaced in 1988 by the Plutonium Fuel Production Facility (PFPF), a fully automated plant with a fabrication capacity sufficient to fuel the Joyo and Monju FBRs. After a few years of operation, PFPF production was temporarily suspended in 1998 (due to a sodium leak in the secondary heat transport system of Monju in 1995). This provided the opportunity to perform planned maintenance and refitting of the facility on the basis of lessons learned, including adapting or replacing equipment which had malfunctioned and reducing the accumulation of plutonium hold-up in the fabrication line. The plan to expand PFPF by adding a production line for ATR fuel has been abandoned with the decision to cancel the intended Ohma demonstration ATR NPP.

In the Russian Federation, three small scale facilities started operation in the 1970s for FBR fuel: the Granat/Chelyabinsk laboratory scale facility and the Paket/Chelyabinsk small scale facility, both fabricating pellet fuels, and the Research Institute of Atomic Reactors (RIAR)/Dimitrovgrad integrated reprocessing–refabrication facility, producing Vipac fuel [39].

Table I summarizes the main characteristics of the MOX manufacturing facilities functioning today. The seven operators of these facilities have an aggregate experience of some 160 years of MOX fuel fabrication. To this should be added the valuable technology background of the now defunct Siemens MOX operations and of the laboratory scale and pilot scale facility predecessors to the facilities listed in Table I. The available capacity for LWR (+ATR) fuel is approximately 210 t HM/a, which is insufficient to serve the short and medium term demand from customers for return of their separated plutonium as MOX fuel; this is one cause of the increasing

TABLE I. CURRENT CHARACTERISTICS OF OPERATING MOX FUEL FABRICATION FACILITIES

Country	Facility	Operator	Start of operation	Capacity (t HM/a)	Feed	Product	Process
Belgium	BN/Dessel	Belgonucleaire	1973	40[a]	PuO_2	LWR FRs	MIMAS
	FBFC Int'l	FBFC	1987[b]	120–200[c]	FRs	LWR FAs	Assembly
France	CFCa	Cogema	1962	10	PuO_2	FBR FAs	COCA
			1989	40[d]	PuO_2	PWR FRs	MIMAS
	MELOX	Cogema	1995	100[a]	PuO_2	PWR FAs	MIMAS
India	Tarapur	BARC	1994	18	PuO_2	BWR FAs	Conventional[e]
Japan	PFFF	JNC	1972	10[f]	$(U–50\%Pu)O_2$	ATR FAs	Conventional
	PFPF	JNC	1988	5[g]	$(U–50\%Pu)O_2$	FBR FAs	Conventional
Russian Federation	Paket	Mayak	1986	0.3	PuO_2	FBR FAs	Conventional
	ERC	RIAR	1981	1	Spent fuel	FBR FAs	Vipac
UK	MDF[h]	BNFL	1994	8	PuO_2	PWR FAs	SBR

[a] Capacity restricted by licensing.

[b] New part of a uranium fuel fabrication plant in operation since 1963.

[c] 120 t HM/a (700 FAs/a) (BWR only); 200 t HM/a (440 FAs/a) (PWR only).

[d] Capacity if no FBR fuel is being fabricated.

[e] 'Conventional' means pellet fabrication processes based on compaction granulation of the feed powders, as is used in most uranium fuel fabrication plants.

[f] The licence is based on plutonium processed annually, i.e. 850 kg Pu/a.

[g] The licence is based on plutonium processed annually, i.e. 2.5 t Pu/a.

[h] The facility ceased commercial operation in 1999 and, in the future, it is intended that it will be used as a technical support facility to the Sellafield MOX plant [40].

plutonium stockpile over the past decade. The available fabrication capacity for FBR fuel is over 20 t HM/a, which significantly exceeds demand [29].

3.1.2. Fabrication processes

The various fabrication processes developed historically and applied currently, well described in some detail in Refs [7, 29], will only be summarized here.

3.1.2.1. Powder processing routes

In enriched uranium, the fissile material is inherently present in the fuel. In MOX fuel, the fissile material, plutonium, has to be added to the carrier material, uranium. This blending of two fissile/fertile materials is the most specific difference between uranium and MOX fuel manufacturing.

The conventional fabrication route is a direct application of the most common industrial fabrication process for uranium fuel, the enrichment of uranium being replaced by a mechanical blending of the feed powders: UO_2 and PuO_2, or $(U–Pu)O_2$ for the plutonium delivered by the JNC reprocessing plant. As the blended powder is not free flowing, and is therefore unsuitable for feeding to a pellet press, the powder is preconditioned by precompaction in a slugging press, followed by granulation, the granules being obtained by crushing the slugs. The challenge in this process is to obtain a uniform distribution of the plutonium in the product [29]. Optimizing the ball (or attritor) mill is of paramount importance for achieving uniformity of the plutonium distribution, as well as a good dispersion of the lubricant and of the pore former, if the use of a pore former is required.

Figure 5 illustrates a schematic flow sheet of such a conventional fuel fabrication process. It represents the process applied by JNC in PFFF and PFPF and, approximately, by Mayak in Paket. While JNC most commonly starts from co-denitrated $(U–50\%\ Pu)O_2$, Paket is fed by PuO_2, as are all other current MOX fuel manufacturing plants.

The elaborate process currently applied by JNC in PFPF (Fig. 6) provides a particular example of a conventional powder production process. The additives utilized include a binder, to improve granulation, a pore former, to achieve the low density specified for FBR fuel, and a lubricant, as universally used to optimize pelletizing. In such processes, de-waxing of the green pellets is required prior to sintering. Excellent quality fuel can be fabricated by such a process.

The BARC plant also utilizes the conventional process, with the specific feature that the blending and ball milling operations are conducted in a single attritor mill, as initiated by BNFL.

A simplification of the conventional feed powder processing route (which is to MOX fuel what the enrichment operation is to uranium fuel) was developed in the

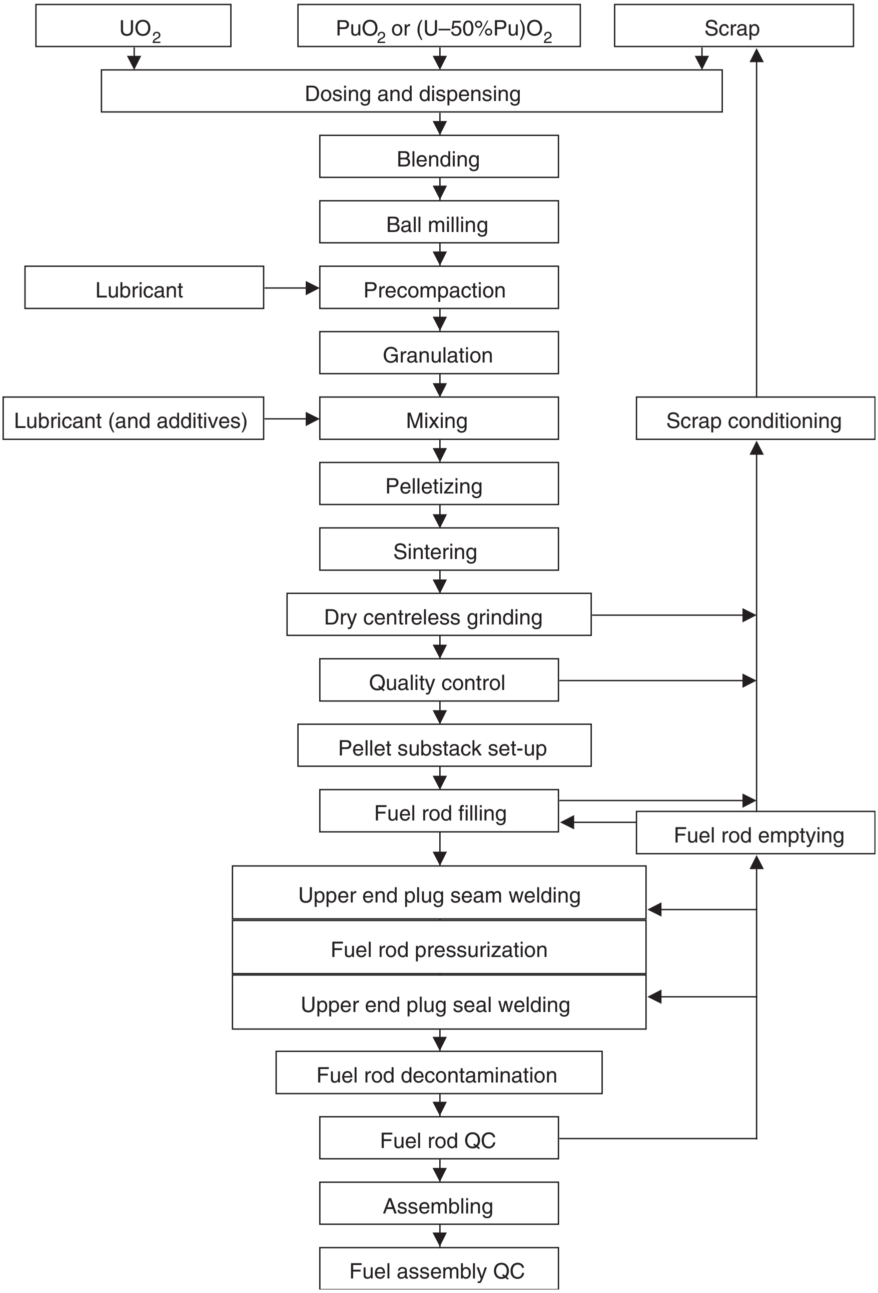

FIG. 5. Flow sheet of a conventional fabrication process.

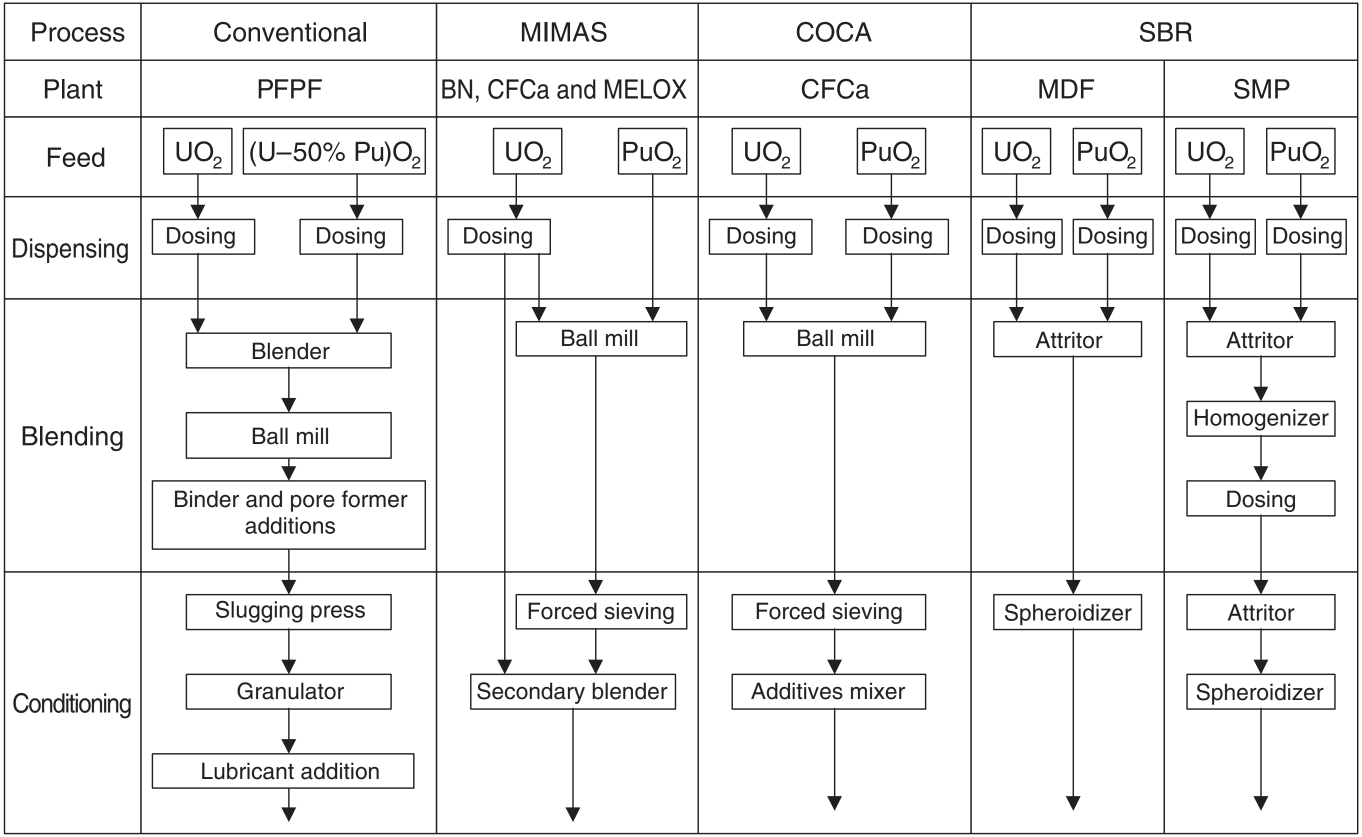

FIG. 6. Schematic flow sheets of the powder processing routes (incorporation of additives and scrap recyling not represented).

1970s by CEA and applied in the Cadarache fabrication plant under the name COCA, an acronym for Cobroyage (co-milling) Cadarache. It is based on the use of an optimized ball mill acting as a blender and of a forced extrusion of the lubricated micronized powder through a sieve, resulting in free flowing granules adequate for feeding the pellet press (Fig. 6). This process, originally developed for FBR fuel, has been adapted for LWR fuel and used from 1989 to 1994 for manufacturing MOX fuel for EDF's PWRs. However, this adaptation has proved to be difficult to master [41] and, as a result, the micronized master blend (MIMAS) process has been preferred for LWR fuels and is now applied in this plant [29].

An original development was conducted in the 1980s by BNFL and resulted in the short binderless route (SBR), based on the application of alternative process equipment for the blending and granulation functions (Fig. 6). The traditional ball mill is replaced by an attritor, an off-the-shelf mill widely used in the pharmaceutical industry and adapted for providing good blends of constituents in a short processing time. The precompaction granulation equipment is replaced by a spheroidizer, working on a powder agglomeration principle [29].

The MIMAS process, invented by Belgonucleaire in the early 1980s, is an adaptation of the reference fabrication process developed earlier and applied commercially in the 1970s at the Dessel plant. The reference process consisted of a single blending of PuO_2 powder with free flowing UO_2 powder, resulting in a blend of adequate flowability to feed the pellet press. However, when the reprocessors decided that even unirradiated MOX fuel had to be almost completely soluble in a pure nitric acid solution, the reference fuel was deemed to be no longer acceptable [29]. To meet this new specification, the earlier single blending step was replaced by a two step blending approach: in the first step, the pure PuO_2 feed and some UO_2 are co-micronized resulting in a master mix of UO_2–(typically)30% PuO_2, which is the fundamental principle of the MIMAS process (Fig. 6); in the second step, the master mix is blended down with free flowing AUC or ADU UO_2 to the specified plutonium content of the MOX fuel. The very close contact between the micronized UO_2 and PuO_2 particles provides for adequate interdiffusion during sintering and therefore the required solubility. In parallel, Alkem (subsequently part of Siemens) had developed and applied commercially the sibling optimized co-milling (OCOM) process [7], with a similar success in fabricability and improved fuel behaviour. The main differences between MIMAS and OCOM were in the ball milling and powder conditioning steps.

While Paket uses a conventional process for manufacturing most of its FBR fuel, one third of the production has used ammonia co-precipitated MOX powder and a further 5% of the production was by a sol–gel process, i.e. two alternative processes developed at Mayak. The facility uses a high energy mixer for milling and blending feed oxides and scrap.

The process being developed at the Experimental Research Complex by the radiochemistry department (ERC/RIAR) there starts with spent fuel and produces

Vipac fuel. It is based on molten salt dissolution in a ‘chlorator–electrolyzer’ made of pyrolitic graphite; the electrorefined UO_2 and/or MOX is deposited on the cathode as a loose crust that is crushed and sized to produce the required size fractions which are fed into the fuel rod by vibro-compaction. More details on the process can be found elsewhere, for example Ref. [36]. Although further development is being pursued, the facility has already produced fuel [37].

3.1.2.2. Fabrication technology

Following preparation of the MOX powder, all the subsequent processes are similar with some minor variations. Pelletizing is carried out in hydraulic presses; only Siemens/Hanau has installed rotary presses in their large plant. Sintering is most commonly conducted in continuous furnaces, except in PFFF, which is equipped with batch furnaces, and in PFPF, currently equipped with both types of furnace. Centreless grinding is now performed dry, which, in principle, does not require subsequent drying of the ground pellets. However, the JNC operation includes a degassing step after sintering for the low density Monju fuel pellets. Tungsten inert gas (TIG) welding is commonly adopted, for both the seam and the seal welds of the end plug. Only Siemens/Hanau used resistance welding, as for their uranium fuel.

For FBR FAs, the assembling operation starts, in most cases, with fitting a helical spacing wire to the rods and involves the introduction of a bundle of FRs into the FA wrapper tube (hexcan). For LWR FAs, the FRs are positioned in magazines and drawn from the magazines through the FA skeleton. For ATR FAs and most of the UK FBR fuel (up to 1988), individual FRs are directly loaded into the gridded FA skeleton.

As for the powder preparation routes described above, each fabrication plant is characterized by specific processing approaches, influenced by their design bases and licensing limits. In this respect it is worth noting, amongst other factors, the minimum ^{240}Pu content of the plutonium to be processed, its maximum americium content, the maximum percentage of plutonium in the MOX fuel and the maximum allowed personnel exposure. These are provided, as illustrations, in Ref. [29] and influence the equipment adopted and the plant layout.

The Belgonucleaire plant has fitted the original MIMAS process to the requirements of their 40 t HM/a industrial operation. An almost uniform isotopic composition throughout a fabrication campaign is achieved by computerized selection of the feeding sequence of PuO_2 supply cans to the process, guaranteeing thereby the energetic equivalence of all the FAs throughout their utilization. The automated operation of a sophisticated ball mill provides for homogeneous and uniform distribution of the PuO_2 in the 60 kg MIMAS master blends. Developments have broadened the types and proportions of scrap that can routinely be dry recycled in the process (Section 3.1.5). The 80 kg capacity of the secondary blender, optimized to obtain a uniform distribution of the master blend in the free flowing UO_2, is a compromise between simplification of product traceability and minimization of the scrap and waste arisings, taking due consideration of the required plant flexibility. In general, all the items of equipment and their sizes have been conceived in line with the objectives of the plant: producing fuel to a large variety of specifications in rather small fabrication campaigns (typically 4–29 t HM, each comprising three to eight different fuel compositions and/or FR types). The FR filling and welding unit is designed with minimum intrusion of the cladding into the glovebox, to minimize FR surface contamination, decontamination and contamination monitoring.

CFCa, later renamed Cogema/Cadarache, which has historically been the largest FBR fuel plant with two dedicated fabrication lines, has acquired and implemented the Belgonucleaire LWR fuel technology to launch industrial LWR operations while, at the same time, testing automated transfer and quality control (QC) devices for MELOX [15]. The adjacent Commissariat à l'énergie atomique (CEA) plutonium facilities provide for scientific research into the process parameters, resulting in an opportunity to improve fabrication continuously, complementing the large database acquired by the past industrial operation of the MIMAS process.

The MELOX plant [32] is the first large scale LWR MOX fuel facility. The MIMAS process was adapted ('A-MIMAS') to achieve the main objective of the plant, namely to fabricate fuel in large fabrication campaigns (typically 90 t HM, each consisting of only three discrete plutonium contents). In this context:

(a) Complete automation has been implemented from the selection and opening of the PuO_2 canisters to the emptying of individually chosen PuO_2 cans.
(b) The ball mill has been maintained at 60 kg capacity, providing the possibility to use up to 70% pellet scraps as input and resulting in the same excellent homogeneity and uniformity of PuO_2 distribution in the A-MIMAS master blend as from the original ball mills.
(c) A high capacity (640 kg) secondary blender, consisting of a conical screw mixer with a double envelope air cooling system, has been adopted.
(d) An automatic video inspection of the pellets with rejection of out-of-specification pellets has been incorporated in the fabrication line.
(e) The filling, welding and decontamination of FRs has been enclosed in one single glovebox, containing essentially the same industrial equipment as used for uranium fuel in the FBFC plants.
(f) The FA manufacture, QC and handling have been fully automated.
(g) The waste is minimized through sorting and treating it and irrecoverable scrap in a dedicated building on the MELOX site, in the liquid effluent treatment unit of Cogema Marcoule and in the centralized UCD and URP facilities at Cogema La Hague.

As its name indicates, MDF was conceived as a test facility for the SBR concept and not as an industrial fabrication facility: the emphasis was to put it rapidly into operation and gain irradiation experience of SBR fuel, rather than to optimize the equipment layout. Attention was mainly exercised on the demonstration of four specific features:

(1) The vertically integrated attritor–spheroidizer system.
(2) The transport system of pellets: picking and placement of the green pellets in the sintering boats and cushion transfer.
(3) The loading system of pellets into the FRs.
(4) The development of all the QC procedures and techniques required for commercial fuel.

The BARC facility, devoted to the development and demonstration of an industrial fabrication technology, is described in Ref. [38].

Similarly, the PFFF and PFPF plants are described respectively in Refs [33, 42]. PFPF has raised the fabrication technology to a very high degree of automation. The experience resulting from the problems encountered (e.g. the historical high level of plutonium hold-up and the jamming of the sintering furnace) has resulted in innovations and is invaluable for progressing towards a proper selection of MOX fabrication equipment and appropriate technology.

The facilities in the Russian Federation are developing technology for future industrial fabrication plants and, in this context, are producing demonstration quantities of MOX fuel [43]. This will define the technologies and equipment appropriate for future deployment [44, 45].

3.1.3. Fabrication records

Tables II and III provide, for LWR and FBR fuels, respectively, the production quantities from all facilities since the start of their operation until 31 December 2000. Although definitively shut down, the Siemens/Hanau plant, the Sellafield FBR facility and the FBR line of PFFF have been included in recognition of their significant contribution to MOX fuel industrialization. In these tables, and throughout this section, only deliveries accepted by the customers are included; additional quantities either rejected or still to be accepted by the customer have, of course, been fabricated. Operators of the current fabrication plants have also fabricated small additional quantities of fuel in the 1960s in earlier laboratory (and sometimes demonstration) facilities: Belgonucleaire SCK/CEN (at Mol), Siemens (at Karlsruhe) and the United Kingdom Atomic Energy Authority (UKAEA).

While the fabrication experiences for both types of fuel are equivalent in terms of numbers of FRs and FAs, the tonnage of fuel produced and of plutonium processed

TABLE II. LWR FUEL FABRICATION RECORDS AS OF 31 DECEMBER 2000 (*rounded figures*)

Facility	Since facility startup				1999 (t HM)	2000 (t HM)
	t HM	t Pu[a]	FRs	FAs		
Belgonucleaire	467	26	246 000	1420[b]	38	38
Siemens/Hanau	158	6.4	77 000	380	—	—
CFCa	248	16	126 000	480[b]	40	41
MELOX	455	24	254 000	1050[c]	104	102
MDF[d]	14[e]	1[e]	7 300[e]	36[e]	Nil	Nil
BARC	3	0.1	800	23	1	1
PFFF (ATR fuel)	120	1.9	22 500	750	Nil	Nil
Total	1500	80	700 000	4100	180	180

[a] Contained in the delivered fuel.

[b] Mainly manufactured at FBFC.

[c] Includes 92 FAs incorporating pellets and FRs fabricated at CFCa.

[d] The philosophy was to make different fuel designs (Table IV) to support business in SMP.

[e] Out of which, respectively 3.9 t HM, 0.3 t Pu, 2112 FRs and 8 FAs were later not accepted by the customer [46].

TABLE III. FBR FUEL FABRICATION RECORDS AS OF 31 DECEMBER 2000 (*rounded figures*)

Facility	Since facility startup				1999 (t HM)	2000 (t HM)
	t HM	t Pu	FRs	FAs		
Belgonucleaire	4.2	1.3	14 000	70	—	—
Siemens/Hanau[a]	5.9	1.9	26 000	100	—	—
CFCa	110	22	430 000	2300	Nil	Nil
Sellafield[b]	13	2.6	98 000	300	—	—
PFFF	4	1.1	44 000	375	—	—
PFPF	10	2.5	63 000	401	Nil	Nil
Paket	1.4	0.5	6 700	53	Nil	Nil
RIAR	4.3	0.5	26 000	436	Negl. [c]	Negl. [c]
Total	150	33	710 000	4000	Negl. [c]	Negl. [c]

[a] Now being decommissioned.

[b] Decommissioned.

[c] Less than 1 t HM.

is larger for LWR fuel than for FBR fuel and the LWR experience is more contemporary.

The fuels mentioned in Tables II and III are for a large variety of NPPs in thirteen countries (Table IV).

The experience over the past decade encompasses a broad range of characteristics [29], covering the full range of industrial MOX fuel of current design. It is particularly important to notice that personnel exposure has been generally decreasing or stabilizing over the years, notwithstanding the increasing radioactivity of the plutonium being processed and the increasing plutonium contents of the fuel being fabricated. Indeed, while plutonium derived from reprocessing of gas cooled reactor (GCR) fuel was a common feed in the 1980s, almost all the feed is now plutonium derived from reprocessing of LWR fuel, with consequentially higher specific gamma and neutron activities.

3.1.4. Fuel quality

As is evidenced in a large number of publications by fabricators as well as customers, the fuel produced today meets the specification requirements and is of a quality equivalent to top grade uranium fuels. A good illustration is the burnup achieved by commercial MOX fuels in NPPs (Section 5).

The difficulties of working in gloveboxes without adequate space in facilities not originally designed for industrial production can lead to short-cuts in the manner in which procedures are actually applied [46]. Owing to the importance of MOX fuel quality to the customers’ licensing authority and the public, any malfunction in the QC and/or QA related systems has serious consequences for the fuel manufacturer [46, 47].

It is beyond the scope of this report to overview all the quality attributes of MOX fuels, FRs and FAs. Various publications have dealt with this topic or have been devoted to it, for example Refs [31, 48]. Only two characteristics, approached and achieved differently by each specific fabrication route, will be reviewed: the homogeneity of the plutonium distribution in the MOX fuel and the uniformity of the plutonium isotopic composition within a fabrication campaign.

3.1.4.1. Homogeneity of plutonium distribution

A homogeneous distribution of the plutonium within the fuel serves four purposes:

(1) It imparts to MOX fuel the same resistance to reactivity initiated accident (RIA) failures as that for uranium fuel. This (historically first) requirement resulted from the Special Power Excursion Reactor Test (SPERT) power burst

TABLE IV. NPPs FOR WHICH THE FUEL HAS BEEN FABRICATED
(status as of 31 December 2000)

Plant		Type[a]	Belgo-nucleaire	Siemens	CFCa	MELOX	Others[b]
Belgium	BR 3	P	x[c]				
	Doel 3	P	+[d]				
	Tihange 2	P	+				
Canada	NPD	H	×				
Czech Republic	Beznau 1	P	+				M
	Beznau 2	P	+	+			M
	Gösgen	P	+				
France	Blayais 1	P				+	
	Blayais 2	P	+		+	+	
	Chinon B 1	P				+	
	CNA	P	×	×			
	Chinon B 2	P				+	
	Chinon B 3	P				+	
	Chinon B 4	P				+	
	Dampierre 1	P	+		+	+	
	Dampierre 2	P	+		+	+	
	Dampierre 3	P				+	
	Dampierre 4	P				+	
	Gravelines 1	P				+	
	Gravelines 2	P				+	
	Gravelines 3	P	+		+	+	
	Gravelines 4	P	+		+	+	
	Phenix	F	+		+		
	St. Laurent B1	P	+		+	+	
	St. Laurent B2	P	+		+	+	
	Superphenix	F			×		
	Tricastin 1	P			+	+	
	Tricastin 2	P			+	+	
	Tricastin 3	P			+	+	
	Tricastin 4	P				+	

TABLE IV. (cont.)

Plant		Type[a]	Belgo-nucleaire	Siemens	CFCa	MELOX	Others[b]
Germany	Grafenrheinfeld	P	+	+	+		
	Isar 2	P			+		
	Philippsburg 2	P	+	+	+		
	Obrigheim	P		+	+		
	Neckarwestheim 1	P		+			
	Neckarwestheim 2	P			+		
	MZFR	H		×			
	KNK	F	×	×			
	Lingen	B		×			
	Brokdorf	P	+	+			
	Unterweser	P	+	+	+		M
	Grohnde	P		+	+		
	Gundremmingen A	B		+			
	Gundremmingen B	B	+	+			
	Gundremmingen C	B	+				
	SNR	F	×	×			
	VAK	B		×			
India	Tarapur 1	B					T
	Tarapur 2	B					T
Italy	Garigliano	B	×				
Japan	Fugen	A					J
	Joyo	F					J
	Monju	F					J
	Takahama[e]	P					M
	Fukushima 1–3	B	+				
	Kashiwazaki Kariwa 3	B	+				
Kazakhstan	BN-350	F					P + R
Netherlands	Dodewaard	B	×				

TABLE IV. (cont.)

Plant		Type[a]	Belgo-nucleaire	Siemens	CFCa	MELOX	Others[b]
Russian Federation	BOR-60	F					R
	BN-600	F					P + R
Sweden	Oskarshamn 1	B	+				
United Kingdom	DFR[f]	F	×				
	PFR[f]	F	×		×		
Totals	PWR + PHWR	38	19	9	17	20	3(M)
	BWR + ATR	13	7	4	—	—	2(T) + 1(J)
	FBR	11	4	2	3	—	2(P) + 3 (R) + 2(J)

[a] A = ATR; B = BWR; F = FBR; H = PHWR; P = PWR.

[b] M = MDF; P = Paket; R = ERC/RIAR; T = BARC; J = PFFF/PFPF.

[c] ×, Reactor now shut down.

[d] +, Reactor still operating (not necessarily with MOX fuel).

[e] Fuel delivered to the NPP, but not loaded [40].

[f] Most of the fuel was fabricated in a BNFL facility now shut down.

experiments [49] conducted in the late 1960s. The conclusion drawn from these experiments was that a fissile particle of some threshold size could potentially pierce the cladding below the fuel failure limit of homogeneous fuel (170 cal/g radially averaged fuel enthalpy). In view of this, the pellet specifications define a maximum plutonium-rich agglomerate size (e.g. corresponding to a pure PuO_2 particle of 400 µm diameter) and a plutonium dispersion criterion (e.g. less than 5% of the plutonium to be present in agglomerates larger than 100 µm). Although more recent Nuclear Safety Research Reactor (NSRR) tests conducted by JAERI have failed to reveal any influence of plutonium-rich particles on the power excursion failure threshold, the MOX specifications continue to include plutonium-rich particle size limits. Modern fuel fabrication technologies can easily meet this specification item.

(2) It minimizes power peaks resulting from plutonium maldistribution. While the pellet-to-pellet enrichment is practically constant in uranium fuel, the plutonium

content of pellets varies in MOX fuel, as a result of mechanical blending of the constituents. The power peaks resulting therefrom penalize the admissible power ratings of MOX fuel. Therefore the pellet-to-pellet homogeneity of the plutonium distribution is included in the specifications. The term ‘macrohomogeneity’ can be used to distinguish it from ‘microhomogeneity’.

(3) It ensures solubility in pure nitric acid solutions. As indicated previously, this requirement was raised in the early 1980s by the industrial reprocessors. As MOX crystallographic lattices containing less than 40–50% plutonium are soluble, the solubility criterion requires that the plutonium content be below the solubility threshold in the individual grains. This plutonium distribution attribute can be called ‘microhomogeneity’.

(4) It minimizes fission gas release (FGR) and the resulting rod internal pressure. This aspect will be elaborated in Section 5.

The degree of macrohomogeneity depends on the sophistication of the blending technology. Great efforts at optimizing the blending equipment and procedure have been made at each fabricator. All other factors being equal, the fabrication routes involving a progressive dilution of PuO_2 into UO_2 can more easily achieve macrohomogeneity than the processes that directly mix PuO_2 and UO_2 to the final required composition. Examples of such progressive blending processes are the MIMAS process, with the intermediate master blend, and the JNC process, with the co-denitrated (U–50% Pu)O_2 feed. Rod scanners are utilized for QC of the macrohomogeneity. In their most sophisticated version, they combine passive and active gamma and neutron scanning and provide for the QC of a large number of other FR attributes.

The degree of microhomogeneity depends on the blending and sintering technologies. In this respect, single step blending fabrication routes are better suited. The achievement of the SBR process is a good example. Through proper optimization of the powder processing and sintering steps, the MIMAS process, which produces generically microheterogeneous fuel, has been improved. As a result, a negligible amount of the plutonium is in agglomerates of a size sufficient to enhance FGR. Various publications have provided examples of plutonium particle size distribution and of their analytical determination, see for example, Refs [50–54].

3.1.4.2. Uniformity of plutonium isotopic composition

Plutonium dioxide is produced by European reprocessing plants in batches of typically 90–100 kg plutonium and conditioned in cans with around 3 kg (Cogema) or 7 kg (BNFL) plutonium for delivery to fabrication plants. While PuO_2 precipitation and finishing is a batch operation at La Hague, resulting in strictly identical isotopic composition of the plutonium throughout the batch, it is a continuous process at

Sellafield, resulting in a ‘rainbow’ transition affecting the first and the last cans of each PuO_2 batch. Depending on the average plutonium content of the MOX fuel, from 3 to 11 PuO_2 batches are involved in the fabrication campaigns typical for most plants but more than 50 PuO_2 batches for MELOX (90 t HM fabrication campaign).

The variability of the plutonium isotopic compositions amongst the PuO_2 batches to be incorporated in a fabrication campaign depends on the variation in types (PWR, BWR, advanced gas cooled reactor (AGR), Magnox) and burnups (from first cores to extended burnup reloads) of fuel that have been reprocessed. With a dwindling number of first cores being reprocessed, the variability of plutonium isotopic compositions is progressively diminishing. However it is still much too large to be neglected, since it critically affects MOX fuel design and performance. To compensate for the variability of the plutonium isotopic composition, Belgonucleaire initiated a plutonium equivalence formulation enabling the plutonium contents of each fabrication campaign, and of each MOX fuel batch within a fabrication campaign, to be corrected relative to the design basis plutonium contents. A similar approach has been adopted by all fuel designers and is now universally applied, except when NPP licence limits also require a consideration of the fissile plutonium content. For FBR fuel, the variability of the plutonium isotopic composition corrected by application of an adequate equivalence formula does not result in deterioration of the fuel quality. For LWR fuel, however sophisticated the equivalence formula, fuel performance is still affected by non-uniformity of the plutonium isotopic composition within a MOX fuel fabrication campaign. For instance, in a PWR, a power peak up to 6% can result from a variability of only 2% ^{239}Pu in the plutonium isotopic composition, even when the plutonium contents are adjusted through a sophisticated equivalence formulation [55]. This illustrates the importance of achieving a uniformity in the plutonium isotopic compositions within a fabrication campaign of LWR fuel.

In the Belgonucleaire plant, uniformity is achieved by computerized selection of the four or five La Hague cans being incorporated in the ball mill to produce one 60 kg master blend. The computer programme takes into account, three months prior to starting up the campaign, the actual characteristics of each PuO_2 can provided (at the reprocessing plant) by the customer, of the PuO_2 cans in the fabrication plant buffer store and of the scrap cans available for incorporation in the primary and/or secondary blend. On the basis of the optimized feeding sequence defined by the computer programme, the PuO_2 transport sequence is organized from the reprocessing to the fabrication plant, the PuO_2 transport canisters are opened at the fabrication plant, and the PuO_2 and scrap cans are withdrawn from the store. This results in good uniformity of the plutonium isotopic composition amongst all the master blends within a fabrication campaign (for which the customer provides 3–15 PuO_2 batches). If this single homogenization step were to be insufficient, the

MIMAS process intrinsically offers the possibility of feeding the secondary blender with different master blends, providing an additional homogenization opportunity. Uniformity of the isotopic composition is likewise achieved in the LWR fuel fabricated at CFCa, since the same MIMAS process and the same ball mill are utilized.

The MELOX plant utilizes the same 60 kg ball mills and a similarly powerful computer programme for the selection of suitable PuO_2 cans. However, the homogenization is enhanced by a buffer store of 30 PuO_2 cans before ball milling and by the incorporation of at least two (and potentially up to eight) different master blends in the 640 kg secondary blender. While all the primary blends already meet the specified tolerances for isotopic uniformity, the variability achieved on the pellets is even smaller.

The other MOX fabrication facilities do not provide for a specific opportunity to cross-blend the PuO_2, or co-denitrated $(U–50\%\ Pu)O_2$, feed batches within the MOX manufacturing process, except for the relatively large scale fuel fabrication at PFPF.

3.1.5. Scraps and wastes

For their plutonium management, all MOX fabrication plants have incorporated sophisticated systems for real time accountancy (see, e.g., Ref. [56]) and for minimization of the plutonium hold-ups within the equipment and gloveboxes.

Scraps are generated by the process itself (e.g. centreless grinding fines or sludge), by the rejects (e.g. non-conforming pellets) and by the surpluses fabricated within a fabrication campaign before switching to the next fabrication campaign. The latter in-line contingency inventories of MOX powder, pellets and FRs are relatively large whenever the campaign size is small. Adequate management of the scraps is an important consideration having economic (fabrication cost) and environmental (personnel exposure and waste generation) impacts. From this perspective, all the MOX fuel manufacturers have developed and/or are examining technologies (including scrap conditioning) to recycle scraps into the process without prejudicing MOX fuel quality. Belgonucleaire can now operate with up to 76% of primary blend consisting of scrap. Their current experience extends to scrap feed being 23% of the final product. In CFCa, the transition from the COCA to the MIMAS process has resulted in the possibility of incorporating up to 50% scrap in the primary blend. Similarly, the A-MIMAS process in MELOX can accommodate 70% rejected pellets (clean scrap) in the master blend. This corresponds, respectively, to approximately 15 and 20% of the total feed. Trials made by BNFL with UO_2 at Springfields and with MOX at Sellafield have supported the development of an SBR flow sheet with recycle of sintered scrap.

Waste arisings originating from plant operation and maintenance, as well as waste due to originate from plant backfitting and ultimate decommissioning, have also received proper attention. Indeed, waste management influences fabrication costs, personnel exposure, licensability and public acceptance. At the Belgonucleaire plant, by identifying and optimizing the waste generating operations and by educating the personnel, plutonium contained in the waste has been reduced to less than 0.1% of the plutonium contained in the delivered MOX fuel [7], notwithstanding the policy of not stripping plutonium from the waste. This illustrates the accomplishments achievable by feedback from lessons learned. At MELOX, licensing authorizations impose a reduction in the radioactivity releases and the plutonium wastage to almost zero: as mentioned in Section 3.1.2, dedicated facilities have been commissioned at the MELOX/Marcoule site and at La Hague to achieve this target. In addition to this effort by all the manufacturing plants to minimize plutonium in the waste streams, improvement programmes are also being pursued to reduce the volumes of each radioactive waste category.

3.2. ISSUES AND CHALLENGES

In line with the evolution observed over the past few years, MOX fuel fabrication will be confronted with increasingly demanding and difficult targets and conditions.

The evolution to more radioactive plutonium feeds will continue, with the processing of aged plutonium stockpiles and the separation of plutonium from higher burnup uranium fuels. It can be assumed that the incentive to reduce personnel exposure will also persist.

With the increased likelihood of degraded plutonium isotopic compositions and the move to higher design burnups of MOX fuels, the plutonium contents will continue to increase (Section 4.4.2), potentially having an impact on personnel exposure and the quality of MOX fuel. The fuel itself may have to be fabricated to tighter specifications to account for the more severe irradiation duties.

On the other hand, weapons grade plutonium (WPu) will have to be manufactured into MOX fuel in the Russian Federation and the USA. The very high ^{239}Pu content of this plutonium simplifies the handling and processing it has undergone, but imposes criticality precautions and may consequently require adaptation of the equipment. The waste minimization objective will continue to be pursued.

In spite of this context, fabrication costs will have to be controlled, as the continuing decline of uranium prices and the fierce competition amongst uranium fuel fabricators affect the competitiveness of MOX fuel.

3.3. FUTURE DEVELOPMENTS

On the basis of lessons learned, the SBR process tested in MDF has been further developed for SMP [57, 58]. This optimized SBR process reduces the number of QC samples required and results in a larger quantity of fuel with an almost uniform plutonium isotopic composition. The following improvements are noted:

(a) In the feed powder receipt and dispensing units installed on top of each of the two identical powder processing columns, a dual PuO_2 feed has been provided: it allows metered aliquots of plutonium of two different isotopic compositions to be incorporated in a MOX powder lot, to homogenize to some extent the isotopic composition within a fabrication campaign.
(b) A homogenizer and a second stage attritor mill have been added (Fig. 6) to bulk together three 50 kg sub-lots from the first stage mill, with the advantage of constituting 150 kg powder lots.
(c) The four continuous furnaces are capable of operating at up to 1750°C.
(d) Each of the two identical FR fabrication and inspection lines is suitable for both PWR and BWR fuel.
(e) The FA facilities consist of one automatic PWR assembly and inspection line and one automatic BWR assembly and inspection line.
(f) The plant, which is fully automatic, is operated from a central control room which has been designed using virtual reality techniques to ensure that it is ergonomically correct.
(g) An export facility has been built adjacent to SMP, to provide facilities for loading FAs into transport containers for shipment to customers.

From the perspective of preparing for the future, JNC has begun the development of a short process that omits the homogeneous blending and granulation steps, in a further cost reduction initiative [59].

The construction of the Japanese MOX plant (J-MOX), to be sited near the Rokkasho-mura reprocessing plant, is scheduled [60] to start in 2004 and to be completed in 2008. The design capacity has been selected to be 130 t/a of MOX fuel, predominantly for BWRs. The MIMAS process has been selected as the basis for the fabrication technology to be implemented.

As a result of its good performance for the fabrication of FBR fuel, the joint French–German–Russian team has selected the COCA process for disposition of Russian WPu into FBR fuel [61] in the planned DEMOX fabrication plant. The MIMAS process has been retained for the WPu to be dispositioned into WWER fuel [61, 62]. The MIMAS process has also been selected by the USA for fabricating their WPu into PWR fuel.

3.4. CONCLUSIONS

The MOX fuel manufacturing industry has reached maturity as a result of the long operating experience of current fabrication plants. Statistically significant evidence demonstrates that industrially produced MOX fuel achieves the same quality level as uranium fuel and meets customer requirements.

The large scale MELOX, PFPF and SMP plants are examples of extending technologies tried out in smaller plants to building and commissioning more advanced and larger facilities. The lessons learned from experience have been instrumental not only in designing and starting up the new facilities but also in backfitting smaller plants which have operated for many years.

In the next few decades, the industry will be confronted by additional challenges, which will again require similar progressive improvements dictated by experience.

4. LWR FUEL ASSEMBLY DESIGN, IN-CORE FUEL MANAGEMENT AND LICENSING

4.1. STATUS OF EXPERIENCE

MOX fuel assemblies are licensed today for a substantial number of commercially operated PWRs and BWRs in Belgium, France, Germany, India and Switzerland. Additionally, Japan, after performing an LWR MOX demonstration programme, now has four reactors currently licensed for MOX usage. The variety of reactors involved in commercial plutonium recycling and the fuel assembly types in use are shown by Table V. It also summarizes the status of experience with neutronic fuel assembly and core design using commercial MOX fuel. The number of MOX fuel assemblies reloaded demonstrates that plutonium recycling in LWRs has reached industrial maturity [63–66].

4.2. QUALIFICATION OF NEUTRONIC FUEL ASSEMBLY AND CORE DESIGN METHODS

Modern low leakage core designs together with the higher enrichments of uranium and MOX fuel assemblies places increasing demands, with respect to accuracy and reliability, on the design codes. For normal applications, qualified design codes prove to be adequate, accurate and efficient. Current nuclear design

TABLE V. MOX RELOADS IN COMMERCIAL PWRs AND BWRs FROM 1981 TO THE END OF 2000

Country, reactor type (FA type)	Number of reactors licensed for MOX	Number of MOX fuel assemblies reloaded	Maximum FA average $Pu_{tot}(Pu_{fiss})$ (%)/ carrier material	Maximum FA burnup at end of cycle (GW·d/(t HM))*
Belgium				
PWR (17 × 17 – 24)	2	96	7.5 (4.9)/U_{tails}	47.7
France				
PWR (17 × 17 – 24)	20	1400	6.7 (4.5)/U_{tails}	40.0
Germany				
PWR (18 × 18 – 24)	2	72	6.9 (4.6)/U_{tails}	43.4
PWR (16 × 16 – 20)	5	504	6.3 (4.2)/U_{tails}	49.0
PWR (15 × 15 – 20)	1	32	4.3 (3.0)/U_{nat}	42.0
PWR (14 × 14 – 16)	1	45	5.6 (3.8)/U_{nat}	37.0
BWR (9 × 9 – 1, 10 × 10 – 9Q)	2	212	5.4 (3.6)/U_{tails}	50.5
India				
BWR (6 × 6 – 1)	2	10		16.0
Switzerland				
PWR (15 × 15 – 20)	1	68	7.3 (4.8)/U_{tails}	51.0
PWR (14 × 14 – 17)	2	152	6.2 (4.1)/U_{tails}	40.0

* Maximum MOX assembly burnup of regular reloads (not lead test assemblies).

methods for both PWR and BWR MOX fuel employ the standard calculation methods used for uranium fuel, although advanced features for MOX fuel applications have been introduced by some vendors, for example updated nuclear cross-section data, corrections derived from colour set calculations or extensions to the nodal core calculation methods. Examples of the levels of accuracy reached with the modelling of MOX fuel assemblies and high plutonium content BWR and PWR cores can be found in Ref. [65].

The conclusion from comparisons of measured and calculated results is that the prediction of key safety parameters for actual cores loaded with MOX fuel assemblies is of the same quality as that for uranium cores. However, taking into account the smaller database of measurements for cores with MOX fuel and specific manufacturing tolerances associated with plutonium concentration and composition, larger uncertainty margins may have to be applied to certain predicted safety parameters in MOX fuel assemblies, for example the linear heat generation rate and the departure from nucleate boiling ratio (DNBR).

4.3. NEUTRONIC FUEL ASSEMBLY DESIGN

4.3.1. General aspects and design targets

The nuclear characteristics of MOX fuel derive from the neutronic properties of the different plutonium isotopes. The η values for thermal neutrons of the fissile plutonium isotopes are higher than that of ^{235}U — both fission and capture cross-sections are about a factor of 2 larger. The absorption cross-sections of the plutonium isotopes ^{239}Pu, ^{240}Pu and ^{242}Pu show strong resonance peaks in the near thermal region. As a result, the neutron spectrum in MOX fuel is hardened, i.e. at the same power level, the thermal neutron flux is much lower than that in uranium fuel.

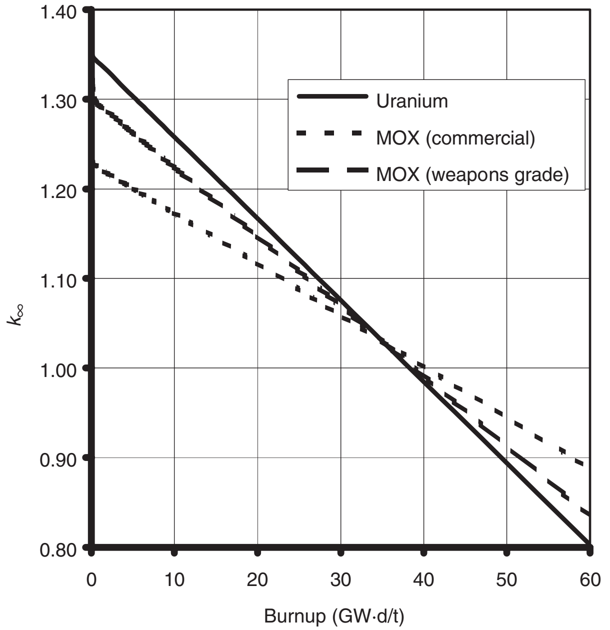

FIG. 7. k_∞ versus burnup for uranium and MOX fuel.

The differences in the neutronic properties of MOX fuel in LWRs result in a slower reactivity (k_∞) decrease with burnup than for uranium fuel (Fig. 7). This figure also indicates that this is particularly true for reactor grade plutonium. In this respect, MOX fuel from weapons grade plutonium behaves much more like uranium fuel. This aspect has to be considered in the process of defining and realizing MOX fuel assembly design goals.

The most common design target for MOX fuel assemblies is lifetime averaged reactivity equivalence to uranium fuel assemblies, i.e. replacing uranium fuel assemblies by MOX assemblies of an 'equivalent' design such that there is no impact on the equilibrium cycle length ('burnup equivalence').

This design criterion is complemented by more general demands. MOX fuel assemblies should be compatible with uranium assemblies in terms of feasible loading strategies and numbers of possible in-core cycles without additional operational constraints, i.e. MOX fuel assemblies have to provide the same operational flexibility as uranium assemblies without sacrificing safety considerations.

In the neutronic MOX fuel assembly design for thermal uranium/MOX cores the large thermal flux gradients at the interfaces between the MOX and uranium fuel assemblies have to be considered. The increasing thermal flux in the direction of an adjacent uranium fuel assembly is addressed by a gradation of the plutonium content of the MOX fuel rods at the fuel assembly edges and corners. Three rod types are typical for the MOX fuel assemblies used in PWRs. Optimized BWR fuel assemblies are more heterogeneous: wider water gaps and larger water structures within a BWR fuel assembly result in MOX fuel assembly designs with up to ten different rod types. Examples of MOX fuel assembly designs are shown in Figs 8 and 9.

For PWRs, MOX fuel assemblies are currently designed without burnable absorbers whereas, for BWRs, neutron poisons are normally required in all fuel bundles. The burnup dependent reactivity behaviour of BWR fuel bundles is optimized by using UO_2–Gd_2O_3 (U–Gd) rods. The feasibility of using U–Gd rods in MOX fuel assemblies has been demonstrated with design studies and by irradiation projects. MOX fuel rods are generally unpoisoned. The expenses associated with development and qualification of MOX–Gd_2O_3 (U–Pu–Gd) fuel rods and the expected excessive fabrication costs for the limited number of gadolinia rods required, using separate manufacturing lines, make U–Pu–Gd designs unattractive.

The 'all plutonium' fuel assembly, comprising MOX fuel rods only, is now commonly adopted for PWRs and BWRs. In comparison with the earlier 'island' type MOX fuel assembly design, which had MOX rods in the centre and uranium rods at the periphery of the assembly, the all plutonium design has clear economic advantages. This is because the additional costs associated with fabrication of MOX fuel assemblies, transport, handling and the back end of the fuel cycle are restricted to a minimized number of fuel assemblies. Note that actual BWR MOX bundles

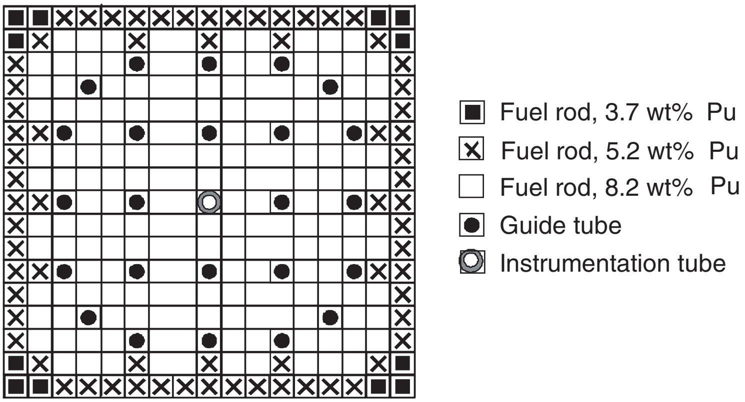

FIG. 8. Example of a PWR MOX fuel assembly design of the 17 × 17 – 24 type with a fuel assembly averaged plutonium concentration of 7.2 wt% Pu.

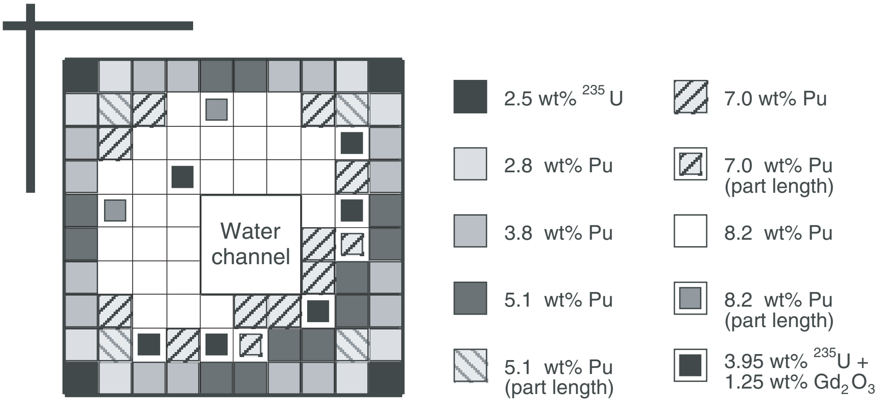

FIG. 9. Example of a BWR MOX fuel assembly design of the 10 × 10 – 9Q type with a fuel assembly averaged plutonium concentration of 5.4 wt% Pu.

(Fig. 9) containing just a few non-MOX (U and U–Gd) rods are still considered to be of the all plutonium type.

Modern MOX fuel assembly designs utilize central water structures, namely water channels for BWRs or additional 'water rods' for PWRs. This additional moderator increases the reactivity of the undermoderated MOX rods and flattens the power density distribution in the fuel assembly without introducing new rod types with higher plutonium concentrations. This design feature became particularly important with increasing plutonium concentrations and burnups. A reasonably flat power density distribution in a uranium fuel assembly environment limits the peak fuel rod burnup within the MOX fuel assembly and is a prerequisite for meeting the mechanical design criteria and for improved operational flexibility. The presence of a centrally located instrumentation tube in some PWR fuel assembly types (Fig. 8) limits, however, the possibility for incorporation of additional water rods.

Finally, studies performed in the past came to the conclusion that the different neutronic characteristics could be accommodated so that no changes in the mechanical and thermohydraulic designs of MOX fuel assemblies were required compared with uranium assemblies loaded in a core at the same time.

4.3.2. MOX carrier material and plutonium composition

The carrier material for the plutonium and the plutonium composition are important parameters for MOX fuel assembly design. Both have impacts on the reactivity characteristics of MOX fuel.

4.3.2.1. Carrier material for plutonium

Three options for carrier material have been investigated for commercial plutonium recycling strategies — natural uranium U_{nat}, tails uranium U_{tails} from the enrichment process and recycled uranium U_{rep} separated during reprocessing of spent uranium assemblies. Early designs were based on natural uranium.

Current MOX fuel is fabricated mainly with tails uranium as the carrier, for economic reasons. Since MOX fuel assemblies are more expensive to fabricate than uranium fuel assemblies, there are economic incentives to concentrate as much plutonium in as few fuel assemblies as possible. Typical ^{235}U enrichments in tails assays are 0.2–0.3 wt%. The somewhat lower ^{235}U enrichment compared with natural uranium can be compensated for by an increase of the plutonium content of 0.4–0.7 wt% dependent on the plutonium quality.

Reprocessed uranium has been used in test fuel assemblies demonstrating its general suitability as a carrier material. From cost considerations, however, its use is less attractive than tails uranium. The isotopic composition of the reprocessed uranium is mainly dependent on the initial enrichment and the discharge burnup of the uranium assemblies. Without further homogenization steps, the variation of the isotopic composition of the reprocessed uranium would be superimposed on the variations in plutonium compositions. This would make MOX fuel assembly design more complex. Industrialized manufacturing processes and monitoring would also necessarily become more demanding. Additional costs for the use of reprocessed uranium would therefore contribute to the already higher MOX fabrication prices.

Furthermore, utilization of reprocessed uranium in MOX fuel would only take care of a small proportion of such arisings and would leave the main stream of reprocessed uranium to be managed in other ways.

4.3.2.2. Plutonium composition

The isotopic composition, or quality, of plutonium is determined by the reactor type in which it is formed, by the initial enrichment of uranium, by the discharge burnup and by the intermediate storage time of the reprocessed fuel. So far, the plutonium used for manufacturing of commercial MOX fuel has been obtained mainly from reprocessing uranium fuel assemblies. Plutonium from the reprocessing of spent MOX assemblies is not yet available in large quantities and has only been used for demonstration purposes. Table VI shows examples of plutonium qualities employed in MOX fuel fabrication. As discussed earlier, variations of the plutonium quality have an impact on the burnup dependent reactivity of the MOX fuel. The higher the plutonium quality the higher is the beginning-of-life reactivity and the larger is the slope of the reactivity versus burnup curve (Fig. 7). This affects the average initial plutonium concentration required for achieving ‘burnup equivalence’ [67]. Total plutonium concentrations of about 7.5 wt% are considered as equivalent to uranium enrichments of 4.0–4.3 wt% ^{235}U for current LWR plutonium.

An aspect taken into account in equivalence considerations for neutronic MOX fuel assembly design is the decay process ^{241}Pu (β decay) to ^{241}Am ($T_{1/2}$ = 14.4 a). The buildup of ^{241}Am starts with the last chemical separation or precipitation step of the plutonium supplied. The loss of the fissionable material (^{241}Pu) combined with the absorption effect of the generated ^{241}Am requires a further increase of the plutonium concentration of the MOX fuel in order to retain burnup equivalence. The lower the plutonium quality the larger is this effect. Current plutonium has a content of 0.7–1.8 wt% ^{241}Am relative to $\mathrm{Pu_{tot}}$ when loaded in a reactor. This is equivalent to a reduction of the effective plutonium concentrations of 0.1–0.2 wt% (0.05–0.1 wt% $\mathrm{Pu_{fiss}}$). The impact of such ^{241}Am concentrations on the burnup equivalence is within the uncertainty limits, but the americium effect becomes important when fresh or partially irradiated MOX fuel needs to be stored for a long time before proceeding with the irradiation. In such cases, the effect can be, and is, accommodated by adapting the core loading pattern.

The fuel assembly design is not sensitive to reasonable changes in the plutonium quality. It is generally based on as-built data with due account taken of variations in the plutonium composition. Revised optimization of the enrichment zones is only required if the recalculation of the fuel assembly design results in unacceptable power density distributions or if a change in the average plutonium concentration is required for safety reasons.

TABLE VI. EXAMPLES OF PLUTONIUM QUALITIES USED FOR MOX FUEL FABRICATION

Quality[a]	Plutonium type (wt%)				
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
Magnox Pu 78.9	0.3	74.3	19.9	4.6	0.9
LWR Pu 68.9	1.5	60.1	24.5	8.8	5.0
Second generation Pu 58.0[b]	1.3	43.8	34.3	14.2	6.4
Weapons grade Pu 95[c]	≈0	95	5	≈0	≈0

[a] Plutonium quality = (^{239}Pu + ^{241}Pu)100/Pu_{tot}.

[b] In a mixed reprocessing regime of MOX and uranium fuel assemblies (Section 7), isotopic plutonium compositions for commercial MOX fuel are expected midway between LWR and second generation plutonium.

[c] So far not used, but fabrication of industrial MOX fuel from weapons grade plutonium is being assessed.

Independent of neutronic aspects related to the core, variations in the plutonium quality and the ^{241}Am content have to be considered for handling and storage of MOX fuel assemblies.

4.4. NEUTRONIC CORE DESIGN

4.4.1. Current status

The operation of nuclear power plants is determined by the demands of electricity suppliers. Load follow operation, variations in cycle length and reload batch size but also unplanned outages are challenges for the flexibility of fuel assembly and core design. MOX fuel assemblies have to provide the required flexibility under such boundary conditions.

Currently, many reactors worldwide are licensed for reload uranium enrichments of 4.0 wt% ^{235}U and higher. In conjunction with low leakage loadings using burnable absorbers, batch averaged burnups of about 50 GW·d/(t HM) in PWRs and 45 GW·d/(t HM) in BWRs have been realized. Increasing plutonium concentrations have to be used in order to match these uranium enrichments.

MOX fuel assemblies with a fuel assembly average plutonium content of about 7.5 wt% in U_{tails} (4.9 wt% Pu_{fiss}) are part of actual reloads for PWRs in Belgium and Switzerland. MOX bundles with a fuel assembly averaged plutonium content of 5.4 wt% in U_{tails} (3.6 wt% Pu_{fiss}) are currently part of regular reloads in two German BWRs. Table VII gives an overview of the status of licensing in commercial plants.

TABLE VII. CURRENT STATUS OF COMMERCIAL MOX LICENSING IN THERMAL REACTORS

Country	Reactors licensed for MOXs (reactor type)	Maximum number of fresh MOX fuel assemblies per reload	MOX fuel assembly content in the core (%)	Maximum[a] Pu_{tot}/Pu_{fiss} (wt%)	Maximum[a] discharge burnup (GW·d/(t HM))
Belgium	2 (PWR)	[b]	24	[b]	50
France	20 (PWR)	16	31	Equivalent to 3.25 wt% ^{235}U	[c]
Germany[d]	9 (PWR)	24	50	[b]/4.65	[e]
	2 (BWR)	68	38	[b]/4.04	
Japan	2 (PWR)	[b]	25	Equivalent to 4.1 wt% ^{235}U	45
	2 (BWR)	[b]	44[d]	Equivalent to 3.0 wt% ^{235}U	40
Switzerland[d]	3 (PWR)	16	40	[b]/4.8	[b]

[a] Fuel assembly average.

[b] No individual restriction.

[c] Limited by maximum of three in-core cycles; no explicit burnup restriction.

[d] Individual licensing situation for each plant type; maximum values provided if applicable.

[e] No general restriction; temporary restriction to fuel rod burnup of 55 MW·d/kg for some plants.

Information on smaller scale activities in Japan and India can be found in Refs [33, 38, 68].

4.4.2. Current trends

As part of an ongoing process, utilities and fuel supplier work on continuously improving fuel utilization. Economic considerations lead to a drive towards an increase in the discharge burnup of spent fuel. Projections for the near future predict batch averaged discharge burnups of at least 55 GW·d/t for PWRs and 50 GW·d/t for BWRs.

In France there is a stated aim to achieve parity between MOX and uranium fuels by 2004. The currently licensed hybrid fuel assembly management, characterized by four irradiation cycles for uranium fuel assemblies (3.7 wt% ^{235}U) and three cycles for MOX assemblies (equivalent to 3.25 wt% ^{235}U), leads to an average discharge burnup of 37 GW·d/t for the MOX fuel and 43 GW·d/t for the uranium fuel. This compares with an average of 45 GW·d/t for the standard uranium quarter core reload management. This approach will be replaced by the 'MOX parity' management scheme. This scheme will comprise an annual quarter core reload with up to 28 of the current standard uranium fuel assemblies and 12 MOX assemblies (equivalent to 3.7% ^{235}U), yielding a maximum fuel assembly discharge burnup of 52 GW·d/(t HM) for both fuel assembly types. The design phase for a gradual introduction of this fuel management approach in the 900 MW PWRs is under way [69].

In general, the trend to higher discharge burnups is accomplished by an increase in the reload enrichment. At a constant cycle length, this has to be accompanied by a reduction of the reload batch size. The limit for ^{235}U enrichment in the fuel fabrication facilities is 5.0 wt%. The highest enriched MOX assemblies actually in use have lifetime reactivities equivalent to uranium fuel assemblies with enrichments of 4.0–4.5 wt% ^{235}U.

At present, the fissile quality of the plutonium returned from the reprocessing facilities is in the range 65–70 wt%. Future plutonium qualities are expected to be from 55 to 65 wt%. Fuel assembly average plutonium concentrations equivalent to ^{235}U enrichments of 5 wt% are 8–10 wt% plutonium with current plutonium qualities. With more degraded plutonium, the average plutonium concentrations have to be increased to 9–13 wt% Pu, with plutonium concentrations in the highest enriched rods significantly above 13 wt% Pu.

Currently, for unmodified lattices, the average plutonium content in a fuel assembly should not exceed 13 wt%. Benchmarking calculations for infinite MOX lattices have indicated that, for plutonium concentrations greater than 13 wt%, k_∞ for voided scenarios may become higher than for fully moderated conditions, leading to a positive void reactivity coefficient. Further investigations are needed for qualification of nuclear data and codes in this respect [70].

4.4.3. Reload strategies

In parallel with the increase in reload enrichments, out–in core loading strategies, where fresh fuel assemblies are positioned at the core periphery, were replaced by low leakage loadings with their significantly improved fuel economy. The low leakage ratio can be defined as the number of reload assemblies not facing the core periphery divided by the total number of reload fuel assemblies. The higher the low leakage ratio, the more fuel assemblies need to be limited in their initial reactivity. This can be achieved either by a split of the reload enrichment or, far more commonly, by the use of burnable absorbers. For PWRs, the burnup dependent reactivity characteristic of MOX fuel based on reactor grade plutonium allows the substitution of fresh uranium fuel assemblies having burnable absorbers with MOX assemblies.

Higher reactivity values, and therefore higher power densities at higher burnups, compared with uranium assemblies, are unfavourable for waterside cladding corrosion and fission gas pressure buildup. These phenomena have to be considered for optimized core loadings. In BWRs, with an adequate fuel bundle design, recycling cores do not require significant adjustments of the reload strategy in comparison with all-uranium core designs [71–74].

The beginning of life reactivity of burnup-equivalent MOX fuel assemblies using WPu is comparable with that of uranium fuel assemblies. Meeting the safety criteria with modern core loading schemes may require either a reduction of the initial plutonium concentration for PWRs and BWRs or the use of burnable absorbers in PWRs.

Recycling cores have to meet the same safety criteria as uranium cores. This has to be demonstrated for each designed core or in-cycle independent safety analyses [75, 76]. The safety criteria are determined for normal operation and operational or design basis transients, for example, main steam line break accidents, loss of coolant accidents (LOCAs), anticipated transients without scram (ATWS) or external events. Some of the neutronic aspects affected by plutonium recycling are discussed in the following sections [77, 78].

4.4.4. MOX impact on normal reactor operation

4.4.4.1. PWRs

The control rod worth in MOX fuel assemblies is smaller than that in uranium fuel assemblies. This effect is most pronounced in fresh fuel but declines with decreasing moderator temperature and increasing burnup. The core design has to account for this. For the currently licensed ratio of MOX assemblies in cores, results from studies have indicated that the influence on the net control rod worth (in the stuck rod configuration) caused by changes in the core loading pattern is often larger than the MOX influence. A tentatively smaller control rod bank worth is repeatedly accompanied by a reduced stuck rod worth, which results in only a slightly changed net control rod worth.

The boron system is used not only for continuous reactivity control under normal operating conditions but also for the long term shutdown of plants. The reduced boron reactivity worth can be compensated for, depending on the number and design of the loaded MOX fuel assemblies, by an increase in the boron concentration in the storage tanks or by an increase in the ^{10}B enrichment in the borated water systems if the solubility of HBO_3 is limiting.

The limits for local power, enthalpy peaking factor $F_{\Delta H}$ or DNBR remain unchanged. In-core or ex-core systems are set to the same limiting values as those for uranium cores.

4.4.4.2. *BWRs*

The impact of MOX fuel assemblies on the control rod system is small for MOX fuelled cores. The thermal flux recovers in the large water gaps between the bundles and results in almost unchanged control rod worths. Hence cores are normally designed with a scattered distribution of MOX fuel assemblies, with only one or two MOX assemblies assigned to each control rod.

4.4.5. MOX impact on transients

4.4.5.1. *Reactor kinetics*

The introduction of MOX fuel leads to a reduced delayed neutron fraction, which influences fast power transients and RIAs. The core power responds more rapidly to perturbations of the reactivity in recycling cores. Part of that effect is compensated for by a slightly more negative Doppler coefficient. The reduction of the delayed neutron fraction β in MOX is mainly due to ^{239}Pu ($\beta = 0.22\%$ for ^{239}Pu compared with 0.67% for ^{235}U). ^{241}Pu has a delayed neutron fraction close to that of ^{235}U. This effect is consequently larger using WPu. Safety analyses are generally performed by assuming a conservatively low fraction of delayed neutrons.

4.4.5.2. *Decay heat power*

Decay heat has an impact on large break LOCAs. Directly after shutdown, the decay heat power is smaller for MOX fuel than that for uranium fuel. With time, the decay heat is increasingly determined by the actinides and consequently, after the decay of the short lived fission products, becomes higher for MOX fuel. The differences in the decay heat are of only secondary importance for the maximum cladding temperatures reached during large break LOCAs.

4.4.5.3. Xenon/samarium

The xenon and samarium worths are smaller in MOX fuel than in uranium fuel. The hazards of xenon induced power oscillations are diminished in MOX fuelled cores.

4.4.5.4. PWRs

More negative moderator temperature coefficients have to be considered in cold shutdown reactivity balances and transients with moderator cooldown, in conjunction with the tendency to a reduced control bank worth and a smaller boron worth.

Fresh MOX and uranium fuel rods of identical designs have the same defect threshold in the case of a potential LOCA. With higher burnups, and therefore higher rod internal pressures, the defect threshold decreases faster for MOX rods than for uranium rods. On the other hand, MOX rods, compared with uranium rods, have a tendency to higher powers and therefore higher fuel and cladding temperatures with increasing burnups. The combination of these two effects, together with a higher transient fission gas release, results in a higher defect probability for MOX fuel rods above a certain burnup level in a LOCA. The contribution of high burnup MOX rods to the defect ratio is very much dependent on the power level of the MOX fuel assemblies and thus on the actual core design.

Two parameters of importance in the case of an RIA, apart from the induced reactivity, are the width of the enthalpy pulse and the cladding conditions. The fastest reactivity changes in PWRs are induced by rod ejection incidents. The highest reactivity and enthalpy values are induced by control rods in low burnt fuel assemblies. Even under conservative assumptions, the enthalpy rise in high burnup MOX rods is not significantly different from that in uranium rods of the same burnup level. The fuel rod conditions of high burnup MOX rods may, however, be somewhat unfavourable. Calculations for recycling cores indicate that, for expected reactivity changes, the maximum cladding temperatures stay below critical values and the resulting pellet centre line temperatures do not reach the MOX melting temperature.

4.4.5.5. BWRs

The more negative moderator temperature and void coefficients affect the cold shutdown reactivity balances, transients with moderator cooldown and transients with coolant pressure increase. MOX fuel with very high concentrations of plutonium may cause a shift to less negative values of the void coefficient.

The impact of MOX on BWR transients is generally small for currently licensed MOX ratios and does not restrict reactor operations.

4.5. FUTURE DEVELOPMENTS

The disposition of surplus WPu has been the topic of several studies in recent years (see, e.g., Refs [79, 80]). The direct approach agreed upon for US material [81] is to use it in operating PWRs. The impact of changes in the plutonium composition on fuel assembly design and core design has been investigated in detail and is well known. Accordingly, the programme is defined in a way that ensures safety requirements and design targets are met.

The Russian Federation plans to establish a closed fuel cycle and to make full use of the available plutonium. This plan involves the recycling of excess military plutonium as MOX fuel, partly in WWERs and partly in FBRs. VVER-1000 offers the potential for the use of MOX fuel assemblies based on reactor grade and weapons grade plutonium. The overall impact on the core behaviour has been investigated and found to be generally comparable to reactors of western design. The problems of industrial plutonium utilization are currently addressed in national programmes and in the framework of international co-operation. The status of both projects is discussed in Ref. [82].

The potential for designing 100% MOX cores for existing LWRs is of interest. Feasibility studies for BWRs [83] and PWRs have come to the conclusion that core designs can be found which meet licensing requirements. Dependent on the reactor design, decreased shutdown margins may require modifications of the shutdown systems. The 100% MOX cores allow for increasing the amount of plutonium under irradiation at a reduced level of heterogeneity of the core. An ABWR to be constructed in Japan will be the first plant with an in-built 100% MOX core capability.

The target of maximum plutonium reduction rates can effectively be achieved by the use of inert matrix fuel in 100% MOX cores. Irradiation tests with Pu–Er–Zr inert matrix fuel are under way [82], and tests are in preparation for thorium based fuel. Conceptual studies have come to the conclusion that the use of inert matrix fuel and thorium based fuel in LWRs exhibits the potential for meeting the targets of increased plutonium reduction rates and improved proliferation resistance. Further qualification of the neutronic and thermal/mechanical layouts and the applied design methods, as well as experimental results, is required before a final conclusion about the feasibility of these alternative concepts is possible and their introduction in commercial reactors is realized (refer to Section 11 for more detailed discussions).

4.6. CONCLUSIONS

In-core fuel management with MOX fuel assemblies is common practice in a large number of nuclear power plants in several countries. Commercial reactors with a ratio of about 30% (up to around 50% in some plants) of MOX fuel assemblies in the core are in operation and meet all applicable safety criteria.

The experience gained from such core loadings confirms the applicability of modern design codes for current plutonium concentrations. The impact of MOX on neutronic design and safety aspects is well understood. This provides the basis for the use of higher plutonium concentrations and, as a consequence, increased discharge burnups. Higher MOX loadings (up to 100%) are being investigated and their feasibility has already been demonstrated.

5. LWR MOX FUEL DESIGN AND PERFORMANCE

5.1. FUEL DESIGN AND SAFETY RELATED CHARACTERISTICS OF MOX FUEL

As indicated in Section 4, MOX fuel is designed to satisfy the same operational and safety criteria as uranium fuel under equivalent conditions [84, 85]. This is reflected in the parallel development of design codes to accommodate the specific characteristics of MOX fuel. MOX fuel assembly design has been universally based on uranium fuel design with only minor modifications relating to the neutronic and thermomechanical properties of the MOX fuel itself.

For design and safety evaluations, the following physical characteristics relating to fuel thermal properties are important:

- melting temperature
- thermal conductivity
- thermal expansion
- thermal creep.

The main concerns for MOX fuel rod behaviour are as follows:

- fuel centre line temperature
- fission gas release
- in-pile densification and swelling
- helium gas accumulation and release
- pellet–cladding mechanical interaction (PCMI) under power ramping.

However, since the plutonium concentration of MOX fuel for LWRs is low and these differences from uranium fuel are relatively small, it is recognized that MOX fuel rod behaviour is very similar to that of standard uranium fuel. No additional problems are apparent with the possible exception of higher gas release and hence an increase of rod internal pressure at high burnup. Because of the neutronic properties of the plutonium isotopes in MOX fuel, the reactivity decreases less rapidly with burnup than in uranium fuel and thus MOX fuel dissipates more power later in its life, releasing more fission gas. In addition, the thermal conductivity of MOX is known to be lower than that of uranium by a few per cent, which may give rise to higher fuel temperatures and thus higher fission gas release. Design changes, such as lowering the initial helium pressure and/or increasing the plenum volume in the rod, are sometimes applied to accommodate this higher gas release [86, 87].

The neutronic design of MOX assemblies is based on the simple principle that the assemblies must be equivalent (or as near as possible) to assemblies containing uranium fuel in terms of dissipated power, total reactivity and accumulated burnup. This equivalence is not completely satisfied because of the differences in neutronic properties of the ^{235}U and ^{239}Pu/^{241}Pu isotopes (Section 4). The design of MOX assemblies has therefore been optimized to reduce the power peaking at the uranium/MOX interface and thus obtain the flattest possible power distribution. This is achieved by zoning the assembly, using different plutonium enrichments in a concentric distribution.

With these exceptions — accommodation of gas release and assembly zoning — the fuel rod and assembly design are essentially unchanged from that of an equivalent uranium assembly. It is generally the case that if a modification is applied to uranium fuel then the same change will be incorporated in MOX fuel after due evaluation of the experience with uranium fuel.

5.2. MOX FUEL PERFORMANCE: EXPERIMENT AND MODELLING

National and international analytical programmes to evaluate the performance of MOX fuel compared with that of uranium fuel have been carried out over a period of 35 years [63]. These programmes, which are still ongoing, are providing the data necessary to compare MOX fuel behaviour with that of uranium, to develop specific MOX fuel performance models and to verify design codes. A wide range of variables has been investigated in these experiments, for example fabrication processes, cladding materials, fuel rod geometries and operating conditions. Post-irradiation examination (PIE) of commercial MOX fuel has also added a lot of data on performance and reliability.

5.2.1. Analytical and irradiation test programmes

5.2.1.1. Belgium

The world's first LWR MOX fuel irradiation began in 1963 in the Belgian PWR BR3; Belgonucleaire manufactured the fuel. From 1963 to 1986 the reactor was used extensively to test and qualify successive MOX fuel types. Feed material characteristics, fabrication processes, ^{235}U and plutonium contents, cladding materials, and rod and assembly geometries were among the parameters tested; unpressurized and pressurized rods were also tested. BR3, which was shut down in 1987, offered a broad range of operating conditions and, together with the test reactors at Petten and Halden, provided an important database in support of LWR MOX utilization.

For more than 18 years Belgonucleaire and SCK/CEN (Mol) have jointly managed a set of international programmes designed to provide fuel validation and licensing data in support of MOX utilization in LWRs [88]. The organizations involved have included fuel designers and manufacturers, research laboratories, utilities and other nuclear service providers. Table VIII lists the most recent of the MOX fuel programmes, which are designed to give information on fuel thermomechanical behaviour.

The PRIMO and DOMO programmes were completed in 1994 and 1996, respectively. Both focused on fission gas release and fuel microstructure at high burnups in PWR and BWR environments. The CALLISTO programme was, in effect, an extension of the PRIMO programme in that selected rods from that experiment were subsequently re-irradiated and ramp tested in the BR2 reactor. The results of these experiments indicated that the behaviour of MOX fuel under transient conditions is equivalent to, or even better than, that of uranium fuel.

TABLE VIII. RECENT INTERNATIONAL TEST IRRADIATION PROGRAMMES

Experiment	Fuel type
PRIMO	PWR
CALLISTO	PWR
DOMO	BWR
FIGARO	PWR
NOK-M109	PWR
NOK-M308	PWR
GERONIMO	BWR

The FIGARO programme was designed to evaluate the thermal behaviour of MOX fuel at a burnup of around 50 MW·d/(kg HM) and to compare the fission gas release threshold of MOX and uranium fuels. Two rods were extracted from an assembly irradiated in the Beznau-1 reactor and the irradiation continued in the Halden test reactor with online instrumentation under PWR conditions. Online pressure measurements indicated that the temperature threshold for fission gas release in MOX is close to that of uranium and the central thermocouple showed that fuel conductivity degradation with burnup follows the same trend in both MOX and uranium.

The NOK-M109 programme took advantage of the rod extraction required for the FIGARO programme and eight additional rods were extracted at the same time. A PIE programme was then proposed which extended the available irradiation performance data of MOX fuel. The fuel was irradiated for five cycles (approximately 50 MW·d/(kg HM)) and the PIE carried out focused on fission gas release; this work has only recently been completed.

The NOK-M308 (PWR) and GERONIMO (BWR) programmes are also designed to extend the MOX fuel performance database. The M308 assembly was irradiated in Beznau-1 to a peak pellet burnup of 58 MW·d/(kg HM) while the fuel in the GERONIMO programme, which is still being irradiated in the Gundremmingen BWR, will reach 65 MW·d/(kg HM) when finally discharged. These international programmes have only recently been established and will involve PIE and ramp testing.

5.2.1.2. Canada

Research and development activities on plutonium containing fuel have been conducted by Atomic Energy of Canada Limited (AECL) at its Chalk River Laboratories (CRL) site since 1960, and they remain a strategic part of AECL's advanced fuel cycle programme [89].

Several fabrication campaigns have been conducted in the Recycle Fuel Fabrication Laboratories (RFFL), producing various types of MOX fuel, which were used for both irradiation and physics testing. Recently, CANDU fuel bundles containing 0.5% plutonium in natural uranium were successfully irradiated in the National Research Universal (NRU) reactor at powers up to 650 W/cm and burnups ranging from 13 to 23 MW·d/(kg HM). Two of the bundles had power histories that bound the normal powers and burnups of natural uranium CANDU fuel. These bundles exhibited sheath strain and fission gas release typical of those observed in similarly operated uranium fuel. Burnup extension above 15 MW·d/(kg HM) only had a small effect on fission gas release.

The irradiation of experimental fuel rods containing weapons grade plutonium will begin in early 2001 in the NRU reactor. The fuel rods were manufactured in the

Russian Federation in the framework of the joint programme PARALLEX (Canada–USA–Russian Federation) [90].

5.2.1.3. Commission of the European Communities

Within the framework of plutonium recycling in the LWR programme, ten studies supported by the Commission of the European Communities (CEC) were conducted by public and private organizations from 1974 to 1986 in order to assess the performance of MOX fuel [91]. Fuel elements, representing various fuel design and fabrication routes, were irradiated in several reactors in the European Community, namely BR3 (Belgium), Dodewaard (Netherlands), Garigliano (Italy), Lingen (Germany) and Centrale nucléaire des Ardennes (CNA, France), as well as in the high flux reactor (HFR) at Petten. The neutronic and thermomechanical behaviour was assessed through PIE. The programme demonstrated the excellent performance of MOX fuel, thus proving the technical feasibility of plutonium recycling in LWRs.

5.2.1.4. France

In France, the PWR MOX feasibility studies performed by Framatome and EDF relied mainly on results from international test irradiation programmes. Of particular value were the programmes supported by the CEC and the PRIMO programme. In addition, some MOX fuel rods were irradiated in the CEA experimental reactor, CAP IV, between 1985 and 1987 (under load following conditions). These programmes, together with the fuel material properties obtained through the FBR programmes, provided physics data to verify the design codes and to develop specific MOX fuel performance models [92].

Since then, the French partners have launched an important national R&D programme and continue to support international projects (e.g. the Belgonucleaire programmes and the OECD Halden Reactor Project (HRP)). The need for experimental tests devoted to an improved understanding of MOX fuel behaviour up to high burnups, in both normal and off-normal conditions, including accident conditions, such as RIAs, has been stressed, mainly relating to thermal properties, fission gas release and mechanical properties. The behaviour of failed MOX rods has also been assessed:

(a) The thermal behaviour of MOX fuel at the beginning of its life has been studied through out-of-pile thermal conductivity measurements and in-pile experiments involving the measurement of centre line temperatures of the fuel rods. The recent out-of-pile data obtained in the CEA laboratories are consistent with the GRIMOX 2 irradiation results [93] and confirm the slightly lower thermal conductivity for MOX fuel.

(b) Feedback on fission gas releases in nominal conditions is constantly enhanced from surveillance programme results and, in addition to these, analytical experiments are performed in order to characterize fission gas release in transient conditions [51]. Studies are underway to examine thoroughly the links between the heterogeneity of the MOX microstructure (size and plutonium content of the (U, Pu)O_2 particles, and the microstructure of the UO_2 matrix) and the mechanism and kinetics of fission gas release.

(c) It has been demonstrated through ramp testing that MOX fuel behaves particularly well from the pellet–cladding interaction, and therefore from the power plant manoeuvrability, standpoint [86]. Power ramp tests have been run on two and three cycle refabricated fuel rods irradiated in the St Laurent B1 reactor. Ramp terminal levels up to 480 W/cm have been reached without cladding failure. Nevertheless, studies are still in progress to understand and characterize this effect, attributed to a higher MOX pellet creep rate than that of UO_2 pellets during power transients. For this purpose, two kinds of test are performed: mechanical property measurements on fresh pellets and in-pile tests devoted to intrinsic uranium and MOX thermomechanical behaviour studies on fresh and irradiated fuels [93].

(d) In-pile densification of MOX fuels is also carefully monitored using specific analytical irradiation and PIE [51].

(e) The behaviour of failed MOX rods during irradiation has been assessed by means of irradiation in a water loop of drilled fresh or pre-irradiated rodlets. The fission product release and kinetics were monitored during the test and the evolution of the fuel microstructure examined through PIE [94].

(f) The evolution of the fuel management strategy in French PWRs with the increase of the fuel discharge burnup and the introduction of MOX fuel created the need for new investigations of fuel behaviour in RIAs resulting from control rod ejection. The Institut de protection et de sûreté nucléaire (IPSN) initiated a research programme in 1992 called the CABRI REP Na programme. The main objectives were to study the potential high burnup effects on uranium fuel behaviour, to analyse the MOX fuel behaviour and to verify the adequacy of, or to modify, the present safety criteria previously defined for lower burnup fuels. Four tests were performed on different burnup MOX fuel rods (Table IX). One resulted in a clad rupture. Though from theoretical physics calculations being at a much higher enthalpy level than expected in a reactor, this rupture gives rise to a suspicion of a fission gas effect enhanced in MOX with regard to uranium, leading to a cladding pressure loading [95, 96]. This will be explored further in later tests.

TABLE IX. THE MOX FUEL TESTS OF THE CABRI REP Na TEST MATRIX

Test	Rod	Pulse (ms)	Energy end of peak (cal/g)[a]	Corrosion (μm)	Result and observation
Na-6 (3/96)	3 cycles 47 GW·d/t	35	126 at 0.66 s 165 at 1.2 s	40	No failure, H_{max} = 148 cal/g Maximum strain: 2.65% FGR: 21.6%
Na-7 (1/97)	4 cycles 55 GW·d/t	40	125 at 0.48 s 175 at 1.2 s	50	Failure, H_f = 120 cal/g Strong flow ejection, pressure peaks of 200–110 bar[b], fuel motion in the lower half-zone
Na-9 (4/97)	2 cycles 28 GW·d/t	34	211 at 0.62 s 241 at 1.2 s	<20	No failure, H_{max} = 210 cal/g Max. strain: 7.3% (mean) FGR: 35%
Na-12 (11/00)	5 cycles 64 GW·d/t	63		80	No failure, H_{max} = 104 cal/g Detailed results and PIE data not yet available

[a] 1 cal = 4.1868 J.

[b] 1 bar = 10^5 Pa.

5.2.1.5. Germany

In Germany, two sets of test irradiation programmes in support of thermal MOX were carried out [63]. The first, in the 1970s, utilized MOX fuel fabricated using a process that resulted in poor homogeneity giving rise to solubility problems. The second set of test irradiations, carried out during the 1980s and early 1990s, concentrated on the irradiation verification of modern MOXs fabricated using the OCOM process [97]. Table X lists the main irradiations carried out under the first phase of test programmes and Table XI summarizes those carried out during the second phase.

Within the tests of this second phase programme, fifteen segmented long fuel rods, each axially composed of seven rodlets, were irradiated for up to four cycles. The rodlets were normal fuel rods reduced in length to match the axial thermal flux shape of HFR (the material testing pool reactor at Petten) and thus allowing a simultaneous power increase of the whole rod. In total, 12 rodlets with modern MOX fuel have been transient tested. MOX fuel manufactured using both the OCOM and AUPuC processes have been included in this programme. This has shown that,

TABLE X. SIEMENS IRRADIATION TEST PROGRAMME FOR GERMAN ‘FORMER’ STANDARD MOX FUEL

Reactor	Scope of work	Description
KWO	Power transients	14 test rodlets Max. powers between 260 and 417 W/cm Rod burnups — 9 to 27 MW·d/(kg HM)
Petten HFR	Power transients	10 rodlets pre-irradiated in KWO Ramp terminal powers — 480 to 560 W/cm Rod burnups — 9 to 32 MW·d/(kg HM)
Halden BWR	Instrumented irradiations to determine fuel temperature and fuel densification	IFA 427, 428

TABLE XI. SIEMENS IRRADIATION TEST PROGRAMME FOR ‘MODERN’ GERMAN MOX FUEL

Beginning of irradiation	Rod/FA number	Type of fuel	Rod burnup MW·d/(kg HM)	Transient testing
1980	Segmented rods	AUPuC	23–39	Petten HFR
1981	Reactor A/FA 1	OCOM/AUPuC	6–42	
1984	Reactor A/FA 2	OCOM	9–49*	Petten HFR
1986	Reactor A/FA 3	OCOM-30 and -15	8–45*	

* Rodlet burnup.

despite different powder properties, the fact that both manufacturing processes were optimized with respect to plutonium homogeneity resulted in comparable behaviour. In addition to the segmented rods, demonstration fuel assemblies were manufactured and extensively characterized before irradiation. Included in one of these assemblies was some experimental fuel which had a reduced plutonium content of 15% in the agglomerates; this was irradiated to a local burnup of 45 MW·d/(kg HM) and was designed to study the influence of plutonium homogeneity on irradiation behaviour [98]. The conclusions were the same as from the French experience.

5.2.1.6. Japan

JNC has developed MOX fuel for thermal reactors over a period of more than 30 years [68]. As a part of this development, JNC conducted various irradiation tests of MOX fuels (Table XII) in thermal reactors such as the Fugen ATR and the Halden boiling water reactor (HBWR).

MOX fuel properties such as fission gas and helium release, microstructure, densification and swelling were comprehensively monitored up to high burnup. These data were useful for the development of the MOX fuel performance code FEMAXI-ATR.

A series of power ramp tests on ATR MOX fuel segments exposed up to 22 MW·d/(kg HM) revealed a failure threshold higher than that reported for UO_2 BWR fuel. ATR MOX fuel rods were also subjected to power cycling irradiation simulating a daily load follow operation. Diameter measurement and fuel instrumentation confirmed that cladding deformation by PCMI, which had occurred at the beginning of the cycling, was immediately relaxed and that there was no mechanical effect caused by repetition of the power change.

Generally, the Japanese industry is strongly involved in the international programmes (e.g. Belgonucleaire, Halden Reactor Project) dealing with MOX fuel [99–102].

5.2.1.7. The OECD Halden reactor project

The OECD HRP has defined an extensive experimental programme related to MOX fuels which is being executed with the objective of providing a performance database similar to that available for uranium fuels [103].

In addition to utilizing fresh MOX fuel and re-instrumented segments from LWR irradiations to high burnup, the concept of inert matrix fuel is being addressed. Irradiation in the Halden BWR (HBWR) is performed in rigs allowing steady state, power ramping and cyclic operation. In-pile data are obtained from instrumentation such as fuel centreline thermocouples, pressure transducers, fuel and cladding elongation detectors, and movable gauges for measuring diametral deformation.

The scope of the overall joint programme on MOX testing includes:

(a) Obtaining data on basic thermal performance from low to high burnup, including assessments of changes of conductivity;
(b) Assessing fission gas release and release kinetics;
(c) Deriving information on fuel swelling and densification through evaluation of temperature data and pressure changes as a function of burnup;
(d) Obtaining data on PCMI behaviour and fuel relaxation capabilities;
(e) Exploring the rod overpressure/clad lift-off effect for high burnup fuels;

TABLE XII. MOX FUEL IRRADIATION TESTS IN THE HBWR AND ATR

Irradiation test	Maximum pellet burnup (MW·d/kg)	Maximum power (W/cm)	Plutonium fissile content (wt%)	Pellet type	MOX powder
HBWR					
IFA-514	56	460	4.6	Hollow and solid	MB[a]
IFA-529	34.7	440	6.0	Solid	MB/MH[b]
IFA-554/555	34.4	560	3.4	Hollow and solid	MB
IFA-565	65	460	4.6	Hollow and solid	MB
Fugen					
DATR type	40.3	445	1.0–2.5	Solid	MH
Segment type	32.6	290	1.5–3.0	Hollow and solid	MH
Gd_2O_3 type	49.2	457	1.5–3.9	Solid	MH
Standard type	24.4	498	0.55–1.56	Solid	MB/MH

[a] MB: 100% PuO_2 powder mechanically blended with UO_2 powder.
[b] MH: 50% PuO_2–UO_2 powder prepared by microwave heating, then mechanically blended with UO_2 powder.

(f) Producing high burnup MOX fuel through continued irradiation in HBWR under PWR conditions and providing performance data (temperature, fission gas release and PCMI) for this high burnup;
(g) Assessing the in-core behaviour of fuel where plutonium is carried in an inert matrix, thus avoiding the generation of new plutonium and allowing a more complete burning.

Salient results are related to the threshold for the onset of significant fission gas release and the relaxation behaviour in a power ramp PCMI situation.

5.2.1.8. United Kingdom

British Nuclear Fuels Ltd (BNFL) is currently involved in a number of in-pile irradiation programmes of SBR MOX fuel that includes both PWR [99] and BWR designs, for example the Halden joint programme experiments [104, 105]. These tests incorporate a large amount of in-pile rod instrumentation designed to determine the thermal, dimensional and fission gas release behaviour of SBR MOX fuel under well controlled conditions. The data from these tests demonstrate the satisfactory in-pile performance of SBR MOX fuel, some to burnups in excess of 80 MW·d/(kg HM).

The most recent in-pile test to be undertaken by BNFL started early in 1999 in the Halden BWR and is designed as a comparative study of the fission product release behaviour of SBR MOX and standard uranium fuel. The experiment is highly instrumented and is providing data on stable and unstable fission gas release, thermal performance, fuel densification, fuel swelling, and pellet cracking and relocation. In particular, confirmation of the lower thermal conductivity of MOX fuel and the similar densification behaviour as a function of temperature and grain size have been obtained.

5.2.1.9. USA

The US Atomic Energy Commission (AEC) began plutonium recycling studies in 1956 [106]. The work was concentrated in two programmes, namely the Plutonium Utilisation Programme, carried out at the Hanford National Laboratory using test reactor irradiations, and the Saxton Programme, which was managed by Westinghouse and demonstrated recycling in a small PWR with MOX loadings, and which started in 1965. US MOX development had, by 1975, progressed to the commercial demonstration stage. However commercial utilization awaited the generic environmental statement on the use of mixed oxides (GESMO), which the US Atomic Energy Commission (USAEC) organized to facilitate the industrial application of plutonium recycle, and the availability of plutonium from commercial reprocessing plants. With President Carter's executive order on non-proliferation in 1977, which postponed indefinitely commercial reprocessing in the USA, all research on MOX fuel was abandoned by 1980.

However, since 1998, the US Department of Energy (DOE) Office of Fissile Materials Disposition has sponsored MOX fuel irradiation experiments to assist in the qualification of MOX fuel derived from surplus weapons usable plutonium. Los Alamos National Laboratory (LANL) staff produced MOX fuel pellets containing 5 wt% PuO_2 from surplus weapons components. Nine capsules produced were irradiated in ATR at the Idaho National Engineering and Environmental Laboratory (INEEL) beginning in February 1998. The original goals of this test included investigation of the effects of any residual impurities on the behaviour of the fuel and cladding. The most often mentioned of these impurities is gallium, which is present in the surplus weapons components.

The post-irradiation examinations of 9 and 21 MW·d/(kg HM) capsules are complete — the irradiation revealed no detrimental effects and gallium migration effects have not proved to be an issue [107].

5.2.2. Commercial irradiation and surveillance programmes

Commercial irradiation of MOX fuel on a large scale began in the mid-1980s, and at present several commercial plants are licensed for MOX usage (Section 4).

Table XIII summarizes the main commercial irradiations of MOX fuel that have been carried out in European reactors since the 1960s.

Post-irradiation examination programmes, including poolside inspection and both non-destructive and destructive hot cell examinations, have been incorporated into MOX utilization studies from the very beginning and the database is now fairly extensive. These programmes are implemented in order to verify the performance of the current reloads, to provide data for design code qualification and to provide irradiated material useful for transient tests or analytical experiments. PIE mainly focuses on lead assemblies irradiated to high burnups and/or under more severe operating conditions. In addition, new fuel pellet designs or processes are the subject of detailed examinations.

In Switzerland, MOX fuel was first loaded into Beznau 1 in 1978, and since then a total of 168 assemblies manufactured by Westinghouse, Siemens, Belgonucleaire and BNFL have been irradiated in three PWRs. The assembly burnups are around 45 MW·d/(kg HM) but, as in the case of Germany and France, individual assemblies have reached levels higher than 50 MW·d/(kg HM).

In Germany a total of 1031 fuel assemblies, manufactured by Siemens and Framatome/Fragema, have been irradiated in LWRs at burnups up to 51 MW·d/(kg HM)

TABLE XIII. PRINCIPAL EUROPEAN NATIONAL EXPERIENCE WITH MOX FUEL IN COMMERCIAL LWRs SINCE THE 1960s

Reactor (type)	Initial year of loading	Total number of assemblies	Maximum assembly burnup achieved (MW·d/(kg HM))
Germany			
BWRs	1966	385	51
PWRs	1972	646	49
France			
PWR (900 MW(e))	1987	1480	51
Belgium			
Doel 3 and Tihange 2 (PWRs)	1995	96	45
Switzerland			
Beznau 1 and 2, and Goesgen (PWRs)	1978	168	54

in the years since 1966. Approximately 200 fuel assemblies with modern MOX fuel have undergone poolside inspection as part of the normal fuel assembly and control rod inspection programmes which are carried out during reactor reloads. In addition, 170 MOX fuel rods have been withdrawn from various assemblies and oxide thickness, length and diameter measurements carried out prior to re-insertion and continued irradiation [63]. Table XIV summarizes the major hot cell campaigns that were carried out to verify the irradiation behaviour of the fuel delivered by Siemens. The first phase concentrated on the irradiation behaviour of the original inhomogeneous fuel. The second phase delivered base irradiation data on the improved fuel; this included data on fuel that had experienced different power histories. The third phase has recently started with MOX fuel irradiated at high power ratings to burnups in excess of 60 MW·d/(kg HM).

The results from these PIE programmes indicate that a heterogeneous plutonium distribution in MOX fuel does not affect the irradiation performance

TABLE XIV. PIE PROGRAMMES ON MOX FUEL DELIVERED BY SIEMENS

Time period	Reactor/ number of rods	Objective	Rod burnup (MW·d/(kg HM))	Result/ comment
1976–1978	A/12	Basic irradiation data for rod design and neutron physics. Former standard fuel	8–37	Rod behaviour comparable to UO_2 in spite of 100% Pu agglomerates
1982–1996	A/12+15 segmented	Basic irradiation data for modern fuel	6–49*	Rod behaviour comparable to UO_2, no significant influence of Pu inhomogeneity
1987–1989	B/4	Data extension for different power histories at increased burnups	35–41	FGR sensitive to the power history of the later cycles
1999–2003	C	Verification of high rated fuel to high burnups	1–4 cycles > 60	Rods under irradiation

* Short rod burnup.

when compared with uranium fuel. This is demonstrated for fission gas release and fuel density:

(a) Most of the MOX fuel fission gas release data match those of uranium; the higher values can be attributed to variations in cladding/fuel gap or differences in irradiation history.
(b) MOX fuel density measurements lie in the middle of the uranium data range.

In France, 1480 assemblies designed by Framatome have been loaded into EDF plants since 1987 and 720 assemblies have successfully completed three irradiation cycles. In Belgium, 96 assemblies of Framatome's design have been loaded into two PWRs and 44 of these have been irradiated for three cycles.

An important part of the French experience was obtained through the examination of the three cycle rods of the first reload supplied by Framatome/Fragema to the St Laurent B1 reactor. Several characterized MOX fuel rods have been withdrawn after each of the three irradiation cycles and then extensively examined in hot cells [86, 91, 108]. These data were related to a burnup range up to about 43 MW·d/(kg HM) and to three different plutonium contents of the MOX assembly. An examination of three cycle fuel rods irradiated under load following conditions in the St Laurent B2 reactor has been also carried out. This examination showed that, from both the waterside corrosion and the rod dimension aspects, the MOX rods behaved similarly to uranium rods. However, the rod puncture data indicated a somewhat higher fractional fission gas release than that for uranium rods. This behaviour is mainly explained by the linear heat rates of the MOX rods which are higher than those of the uranium rods at high burnups, but also, to a lesser extent, by the specific MOX fuel material properties. The MOX rods operated in load following conditions behaved similarly to the reference ones operated in base load.

From the beginning of plutonium recycling in France, EDF's objective has been to burn the MOX fuel at the same burnup ratio as uranium fuel. Further acquisition of rod global behaviour as well as of fission gas release data was required at higher burnups in order to achieve the objective of quarter core reload fuel management with a maximum assembly discharge burnup of 52 MW·d/(kg HM) ('UO_2–MOX parity'). In this way, MOX assemblies have been irradiated for an additional fourth and fifth cycles in the Gravelines 4 and in the Dampierre 2 reactors (up to average rod burnups of 53 and 60 MW·d/(kg HM), respectively).

Poolside examinations performed on assemblies irradiated up to five cycles did not reveal any abnormal mechanical behaviour and confirm the general good condition of the MOX assembly structure.

The hot cell data, obtained on the four cycle fuel rods, did not show any burnup enhancement of fission gas release, owing to the relatively low heat rates during the last irradiation cycle: thus any fission gas release enhancement must be attributed to the end of life power rating rather than being solely a burnup effect. The fission gas release data of the commercial fuel rods are summarized in Fig. 10 and compared with some uranium data. PIE on the five cycle rods is still underway.

The surveillance programme also involves examinations of different fuel types. One objective, for instance, is aimed at providing experience on the behaviour of MOX pellets fabricated according to the MIMAS process with two different UO_2 powders, each resulting in a different fuel matrix. The reference UO_2 powder used by the Belgonucleaire plant is from an AUC route. But Belgonucleaire also tested MIMAS fabrication with an alternative UO_2 powder, produced by an ADU route. This is the powder adopted for the operation of MELOX. In total, more than 55 commercial MOX fuel rods have been withdrawn and sent to hot cells for PIE.

BNFL reported the results from PIE of the first SBR MOX fuel fabricated in MDF for a commercial reactor [109, 110]. After three cycles of irradiation, seven of these fuel rods (burnup of 33 MW·d/(kg HM)) were withdrawn and sent for PIE. Non-destructive examinations (NDEs) showed that fuel performance was generally in line with that expected from uranium fuel which experienced the same power history. Rod puncture results showed that cladding creep-down was primarily responsible for the increase in rod internal pressure and that fission gas release was low. Destructive examinations have helped explain the observations of NDE and rod puncture by showing that the underlying microstructure of SBR MOX is primarily a solid solution of UO_2/PuO_2, and

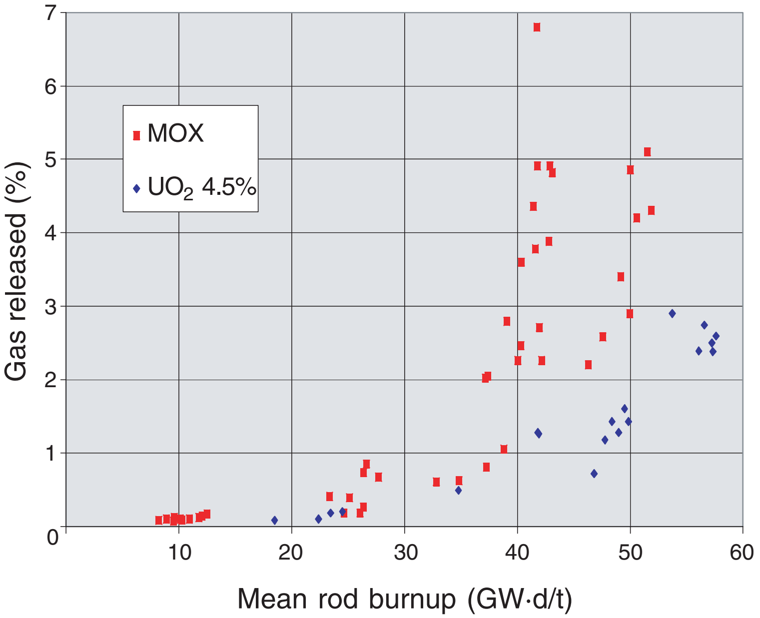

FIG. 10. Fission gas release data of commercial fuel rods irradiated in EDF PWRs.

thus burnup and fission products are evenly distributed throughout the fuel in common with uranium. Power ramp tests on SBR MOX fuel are now under way.

5.2.3. MOX fuel performance and modelling

It has been concluded from both analytical and commercial irradiations that:

(a) The reliability of MOX fuel remains as good as that of uranium fuel — no rod has ever failed for MOX specific reasons [108, 111].

(b) Overall the performance of MOX fuel has proved to be similar to that of equivalent uranium fuel but with a somewhat higher fission gas release (which can be accommodated by a suitable rod design) and with the added benefit of improved resistance to pellet–cladding interaction.

Because of the extremely large uranium fuel performance database and the corresponding well validated codes, it is important to define differences in structure and performance of MOX and uranium fuels. The IAEA Technical Committee Meeting (TCM) held at Windermere in 2000 on Nuclear Fuel Behaviour Modelling at High Burnup and its Experimental Support [112] reaffirmed that fuel performance models developed for uranium fuel and based on uranium experience are generally applicable to MOX. Only a few models need to be adapted to cope with quantifiable differences between MOX and uranium fuels, and only some of these characteristics are affected by the microstructure of the MOX fuel. It has been indicated that MOX fuel modelling has now reached a high degree of development based on a good understanding of the differences between MOX and uranium fuels. Significant progress has been reported with codes capable of evaluating and predicting MOX fuel performance including CARO-E (Siemens [113]), COMETHE (Belgonucleaire [88]), COPERNIC (Framatome [114]), COSMOS (Korean Atomic Energy Research Institute (KAERI) [54]), ENIGMA-B (BNFL [115]), FEMAXI-ATR (JNC [68]) and FPAC (NFI [116]).

5.2.3.1. Radial power and burnup profiles

The neutronic properties of MOX fuel are responsible for a decrease in reactivity with burnup that is markedly less pronounced than that in uranium fuel. This leads to higher power densities at higher burnups in comparison with uranium assemblies. Moreover, the presence of plutonium in the pellet increases the thermal flux depression towards the centre, due to the large absorption of thermal neutrons by the plutonium isotopes. At the beginning of the MOX fuel life the radial power profile is therefore more depressed than that in uranium fuel, whereas at end of life the higher residual plutonium content at the centre leads to higher local powers in MOX fuel [88].

5.2.3.2. *Thermal properties*

Recent reviews of the thermal conductivity of MOX fuel [117–119] conclude that the presence of small quantities of PuO_2 in UO_2 slightly decreases the thermal conductivity. However, there is no agreement about the quantitative trend of the degradation, as has been noted at the above mentioned IAEA TCM on Nuclear Fuel Behaviour Modelling.

For example, KAERI [111] describes this degradation as a two phase material with different thermal conductivities in the matrix and the plutonium-rich agglomerates (the reduction of thermal conductivity ranges from 7 to 10%). BNFL [115] applies a uniform degradation of 8%, while Framatome and Belgonucleaire apply a linear degradation with the plutonium content [88, 114]. In-pile measurements of the centreline temperature of MOX fuel clearly indicate the slight degradation of thermal conductivity of fresh fuel [93, 103]. The French Grimox 2 experiment gave evidence of a 6% higher centreline temperature at the beginning of life (BOL) in MOX fuel compared with uranium for identical operating power levels (Fig. 11). This general result needs to be analysed in detail considering the different contributions to the thermal response.

Measurement of the fuel centre temperature of MOX irradiated to 50 MW·d/(kg HM) showed that the thermal conductivity degradation with burnup in MOX is the same as that in uranium fuel [88]. The melting point of MOX fuel decreases slowly with the plutonium content and with the deviation of the oxygen to

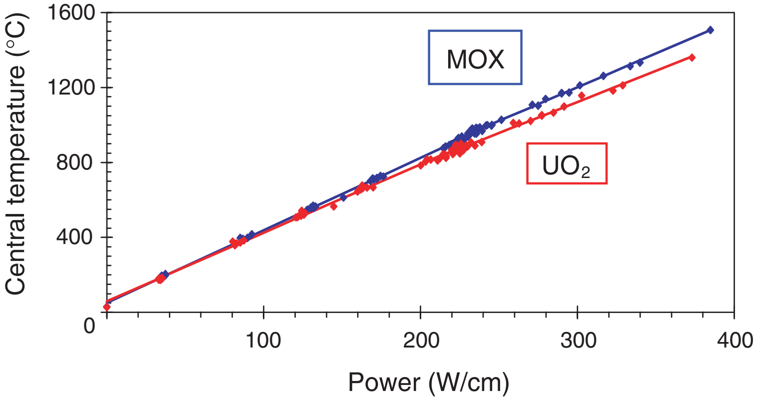

FIG. 11. Centreline temperature of uranium and MOX fuel versus power in Grimox 2.

metal (O/M) ratio. The published data, reviewed recently [118], are in good agreement. There are virtually no differences in the measured thermal expansions of uranium and MOX fuel [88].

5.2.3.3. Fission gas release and fuel microstructure

The complexity of MOX fuel microstructure, both before and during irradiation, and the higher centreline temperature of MOX fuel in comparison with uranium fuel make it difficult to have a simple view of fission gas release from these fuels for the whole range of heat rates, burnups and fabrication technologies [120].

In experimental programmes carried out in the Halden reactor, in-pile pressure measurements performed at different burnups showed that the empirical threshold for significant fission gas release (>1%) derived from uranium fuel also applies well to MOX fuel [8, 103]. However, the fission gas release of MOX fuels with plutonium-rich agglomerates can be expected to differ from that of uranium fuel because of the concentration of burnup and fission products in the agglomerates. The typical evolution of these agglomerates during irradiation is illustrated in Fig. 12.

At the pellet edge, the plutonium-rich agglomerates are well identified by a dense pore population resulting from fission gas generation and bubble coalescence (like the high burnup uranium fuel ‘RIM’ structure). Fine metallic precipitates are also observable. Towards the pellet centre, the bubble sizes increase with temperature. In the agglomerates situated in the central region there is often one large cavity and a small number of bubbles surrounded by large metallic precipitates.

Nevertheless, in MIMAS fuel only the large agglomerates (>30 μm) keep the greater part of the fission products and exhibit this special feature. It should also be

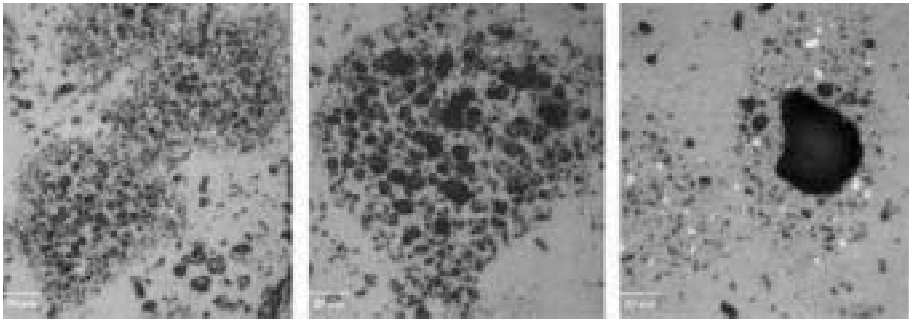
Periphery Midradius Centreline

FIG. 12. Optical micrograph of plutonium-rich agglomerates after 55 MW·d/(kg HM) [121].

stressed that about 25% of the total plutonium content of the pellet is present in these large agglomerates (typical value for a 6 wt% PuO_2 average pellet content). The effect of fission product recoils, 7–9 μm away from their points of creation, as well as plutonium breeding in the UO_2 matrix, is to tend to smooth out the inhomogeneities of the plutonium distribution during irradiation [86, 121].

However, it should be noticed that the difference in fission gas release between MOX fuel and uranium fuel decreases when the homogeneity of the plutonium distribution is improved (as shown by comparison of MIMAS ADU and AUC MOX fuels [121]). This has also been confirmed by the low fission gas release from irradiated MOX fuels with good homogeneity of the plutonium distribution [109, 121].

The use of existing uranium fuel fission gas release models for MOX fuel applications has been reported to give satisfactory results for fission gas release and rod pressure evaluations [88], although some fuel designers are developing, or have developed, MOX specific semiempirical or mechanistic models [54, 114, 122].

5.2.3.4. Operational transient behaviour and fuel creep

Power ramp tests on MOX fuel rods (up to 50 MW·d/(kg HM)) indicated that their integrity was maintained without defects up to power levels higher than 500 W/cm. These results would imply better PCMI performance of the MOX fuel rods than that of the uranium rods [86, 123]. The fission gas release from MOX fuel in operational transient conditions is comparable with that from uranium fuel [86, 87].

Such a favourable PCMI behaviour of MOX is attributed to the higher pellet creep compared with UO_2 in transient conditions. This was demonstrated through PIE of MOX rods after ramp testing [86] and through analytical experiments under irradiation of fuel segments instrumented with cladding elongation detectors or diameter gauges [92, 103].

5.2.3.5. Helium generation and release

Helium is generated in fuel matrices by alpha decay of transuranium nuclides and ternary fission, the former being the major source. Therefore, helium accumulation in the matrices, especially in MOX fuels (about four times that in uranium fuels), increases exponentially with burnup [124]. Studies of helium behaviour showed that the helium diffusion coefficient in UO_2 is several orders of magnitude larger than that of noble fission gases and also that helium is highly soluble in UO_2 [125, 126]. Helium is not released from the fuel as long as its content does not exceed the solubility limit, which depends on the helium partial pressure in the rod. Hence no helium release is observed in pre-pressurized MOX rods (>20 bar) over a broad range of burnups and operating conditions [86, 88]. However, significant helium release is observed in low pre-pressurized BWR MOX fuel rods [124].

5.2.3.6. Defective fuel rod behaviour

The behaviour of defective MOX fuel rods has been monitored through analytical experiments [94] and surveillance of the coolant activity [111, 127]. The release rates of fission products in the primary coolant are similar to those observed with defective uranium fuel. However, careful analysis of the activity ratio of fission products in the primary coolant (or during assembly sipping) allows a MOX failure to be differentiated from a uranium failure.

5.3. MEDIUM AND LONG TERM DEVELOPMENTS

For economic reasons it is clear that MOX fuel should have the same performance as uranium fuel with regard to burnup and operational flexibility. The burnup equivalence between current MOX and uranium fuel assemblies has already been demonstrated in Belgium, Germany and Switzerland. This equality between MOX and uranium products must be maintained on a long term basis and the management of MOX must develop in the same way as that of uranium.

In France, for example, there is a firm commitment to reach ‘MOX parity’ by 2004 (quarter core fuel management, 52 MW·d/(kg HM) maximum discharge assembly burnup) [128]. Moreover, a programme is underway to develop a MOX fuel capable of reaching assembly burnups up to 70 MW·d/(kg HM) over the next ten years [108]. To reach this latter objective, the MOX fuel will benefit from the new rod design and assembly structure developed for uranium fuel, providing margins to internal pressure, and from an optimized $(U, Pu)O_2$ microstructure allowing higher fission product retention.

Most of the fuel vendors, utilities and research centres involved in the MOX industry are pursuing research and development programmes in the following domains in order to improve the performance and to increase the operational margins of MOX fuel:

(a) Acquisition of data from high burnup fuel irradiated in normal and off-normal conditions.

(b) Understanding of the fission gas release and swelling mechanisms in relation to the fuel microstructure and/or the presence of plutonium for modelling purposes.

(c) Acquisition of data from MOX fuel with higher plutonium contents — higher burnups and the use of plutonium from reprocessing of highly irradiated fuel will lead to increases in the plutonium enrichment of pellets.

(d) The impact of helium generation and release on the performance of high burnup fuel. This effect is also important for medium and long term storage, given that helium generation continues after irradiation [129].

(e) Understanding of the role of the microstructure and/or the presence of plutonium on the RIA behaviour.
(f) Development of new microstructures, following initiatives being pursued with uranium fuel using large grains (with or without additives) in order to increase fission product retention at high burnup.
(g) Study of advanced plutonium containing fuels (Section 11).

5.4. CONCLUSIONS

The fuel rod and assembly design for MOX fuel has been universally based on uranium fuel design, with only two minor modifications due to the neutronic properties (addressed by assembly zoning) and to the higher fission gas release (addressed by redimensioning the plenum volume). It is generally the case that if a modification is applied to uranium fuel then the same change will be incorporated in MOX fuel after due evaluation of the experience with uranium fuel.

National and collaborative test irradiations on LWR MOX fuel have been carried out in normal and in transient conditions over a period of 35 years. These programmes, which are still ongoing, are providing the data necessary to compare MOX fuel behaviour with that of uranium fuel, and to enable development of specific MOX fuel performance models and verification of thermomechanical design codes. A wide range of variables has been investigated in these test irradiations, for example, fabrication processes, cladding materials, rod geometries and operating conditions.

In the same period, over 2000 MOX fuel assemblies have been irradiated in commercial PWRs and BWRs with failure statistics indicating that the reliability of MOX fuel is at least as good as that of uranium fuel.

The PIE programmes accompanying both the test programmes and the commercial irradiations all indicate that the overall performance of MOX fuel has proved to be similar to that of equivalent uranium fuel with the added benefit of improved resistance to PCMI.

R&D programmes are still continuing on MOX fuel, as they are for uranium fuel, with the aim of enhancing fuel performance in terms of discharge burnup without penalizing operational and safety margins.

6. TRANSPORTATION

6.1. INTRODUCTION

Since the first loading of MOX fuel into a commercial German LWR (1972), MOX transport activities have undergone significant changes in order to:

- Respond to the increasing number of reactors loaded with MOX fuel,
- Take into account modifications in fuel design and increases in fissile content,
- Adapt the packaging design to applicable IAEA and national safety regulations [130].

6.2. REGULATORY REQUIREMENTS

The objective [131] of the IAEA “Regulations for the Safe Transport of Radioactive Material” [132] is to protect persons, property and the environment. This protection is achieved by requiring under all foreseeable conditions:

— containment of the radioactive contents
— control of external radiation levels
— prevention of criticality
— prevention of damage caused by heat.

All packages and transport operations must comply with applicable national and international transport safety laws and regulations which are, in practically all cases, based on the transport safety regulations recommended by the IAEA. In addition to having been widely incorporated into national laws and regulations, the IAEA recommendations have also been introduced into international regulations including the:

(a) UN Committee of Experts on the Transport of Dangerous Goods — Recommendations on the Transport of Dangerous Goods — Model Regulations [133],
(b) International Civil Aviation Organization (ICAO) — Technical Instructions for the Safe Transport of Dangerous Goods by Air [134],
(c) International Air Transport Association (IATA) — Dangerous Goods Regulations [135],
(d) International Maritime Organization (IMO) — International Maritime Dangerous Goods Code [136],

(e) ADR — European Agreement Concerning the International Carriage of Dangerous Goods by Road [137],
(f) RID — Regulations Concerning the International Carriage of Dangerous Goods by Rail [138],
(g) ADN — European Provisions Concerning the International Carriage of Dangerous Goods by Inland Waterway [139].

In addition, for maritime transport, the 'International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes on Board Ships' (the INF Code) [140] is typically applied to all transport covered by this code — it became mandatory on 1 January 2001.

Shipments of plutonium and MOX have to fulfil the highest physical security requirements. Physical protection requirements are basically laid down in the IAEA document INFCIRC 225/Rev.4 [141] and are further detailed in national regulations and guidelines which are classified as 'restricted'. For MOX shipments, a large number of technical and administrative requirements have to be fulfilled.

The Regulations are part of a framework containing other guidelines and conventions that have a bearing on transport [131]. Guidelines for protecting nuclear material against sabotage and theft are given in the IAEA reports on the Physical Protection of Nuclear Material [141] and in the Convention on the Physical Protection of Nuclear Material [142]. The Convention concerns specifically the international transport of nuclear material. International co-operation is essential when countries are affected by transport accidents. Such accidents or incidents occurring in international waters or air space will be of worldwide interest. Additionally, events occurring within national borders can have implications for neighbouring countries. In recognition of these concerns the IAEA has prepared two conventions: the Convention on Early Notification of a Nuclear Accident [143] and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency [144]. In the event of an accident, the question of liability is covered by two basic international regimes on nuclear third party liability. These are the so-called Paris [145] and Vienna [146] conventions, which are linked via a Joint Protocol [147].

6.3. CURRENT STATUS OF MOX FUEL TRANSPORT

The main products included in plutonium material transport activities are: PuO_2 powder, fresh MOX rods, fresh MOX assemblies and spent MOX rods and assemblies.

Transportation of these plutonium containing products (fresh and spent) is carried out using exclusive packaging types designated 'Type B' and 'Fissile'. Currently a series of flasks (see below) developed and put in service over the last two decades is used.

Each type of package and design must satisfy the requirements of IAEA regulations concerning criticality, shielding efficiency, structural strength, heat resistance and leak tightness under both normal and accident transport conditions. The conformity to safety criteria is demonstrated by calculations and/or by tests conducted on prototype models up to full scale (mainly for drop tests and fire tests).

Transportation is under governmental control and is subject to domestic laws.

6.3.1. PuO_2 powder

If the fabrication plant is not co-located with the plutonium storage facility, PuO_2 powder has to be transported from the reprocessing plant to MOX fabrication plants within B(U)F packaging.

In France, the packaging most commonly used is the FS-47 (Fig. 13), which can accommodate up to 19 kg of PuO_2 powder and is transported by road, with each truck having a payload of ten packagings. Its design allows remote control operations such as loading or unloading and lid screwing or unscrewing.

The current equivalent UK packaging for PuO_2 powder is the '2816 package' (known as the SAFKEG) that can carry up to 18 kg of PuO_2 powder. The packages

FIG. 13. The FS-47 PuO_2 packaging.

are transported in groups of six within stillages. Depending upon the type of high security vehicle employed, a number of stillages can be transported (Fig. 14). The packages come in a number of versions to cater for different plutonium can sizes [29, 148].

6.3.2. Fresh MOX rods

Whenever the FA plant is separate from the MOX plant, there is a need for fresh MOX fuel rods to be transported between the two. In France, for example, the rods are first bundled into steel boxes; each box contains a maximum of 314 PWR rods or 188 BWR rods. The PWR rod boxes are then loaded horizontally into FS-65 1300 B(U)F type packagings (Fig. 15) fitted with a dedicated basket (see Section 6.3.3 for a description of the packaging). The payload is four FS-65 1300 packagings per shipment.

The main traffic for fuel rods is between the Cogema Cadarache MOX plant in France and the FBFC International/Dessel assembly plant in Belgium. Twelve such shipments were performed in 2000.

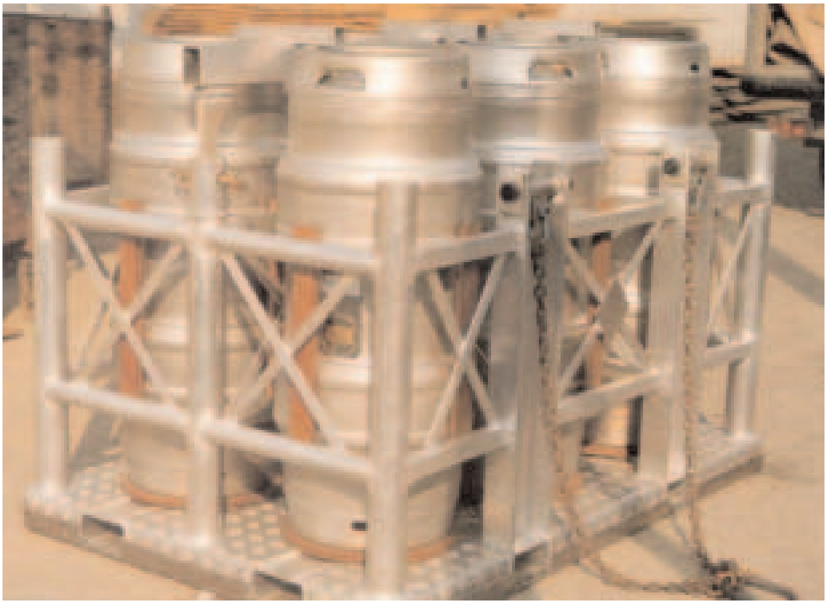

FIG. 14. The 2816 SAFKEGS in a stillage.

FIG. 15. The FS-65 1300 package.

6.3.3. MOX fresh fuel assemblies

For the transport of fresh fuel assemblies, it is particularly important (as for uranium fuel) to ensure that the fuel assembly is not subjected to any unacceptable shock loads or vibration. Continuous recordings of vibration and shock loads are taken using special equipment such as recording accelerometers.

Fresh MOX FAs are transported between assembly plants and power plants in B(U or M)F packagings.

6.3.3.1. European power plants

A number of packages have been utilized and continue to be in service:

(a) The FS-69 (17 × 17, 15 × 15 and 14 × 14) package licensed by Transnucléaire Paris (TNP) in France and the equivalent TNB-176 licensed by Transnubel (TNB) in Belgium (17 × 17, 15 × 15 and 14 × 14). This package accommodates two PWR FAs and is transported by road with a payload of four packages (Fig. 16). Twenty four shipments were made in 2000 between the MELOX plant and the EDF MOX plants. BNFL with TNB have performed a number of air shipments of MOX to Switzerland using the TNB-176.

FIG. 16. The FS-69 package (opened for maintenance).

(b) The Siemens Type Ve package, accommodating two PWR FAs, combined with the MOX outer package (D4295), has been used for transporting MOX 14 × 14 PWR FAs to Switzerland.

(c) The Siemens III package is used to transport MOX FAs to power plants in Germany and Switzerland which use 15 × 15, 16 × 16 and 18 × 18 PWR fuel arrays. This package accepts two PWR FAs and is transported with a payload of two packages per high security vehicle. Thirty-three shipments were performed in 2000 between FBFC International at Dessel and German power plants, and one shipment took place from BNFL to a German power plant.

(d) The FS-41 package is used for delivery of FBR fuels to the Phénix power plant in France.

6.3.3.2. Japanese power plants

(a) For fuels manufactured in Europe

The transport of fresh MOX PWR FAs to Japan from the UK is performed by rail from the Sellafield assembly plants to a UK port and thence to Japan aboard purpose designed Pacific Nuclear Transport Limited (PNTL) vessels.

Fresh MOX PWR and BWR FAs are transported to Japan from France, first by road, from the MOX assembly plant to Cogema La Hague where they are transferred to maritime packagings. These are then transferred to Cherbourg harbour and loaded onto PNTL ships.

FS-65 packages are used for road transport. The FS-65 package was designed to transport MOX rods or FAs of any PWR or BWR design [149]. They are composed of:

(i) A cylindrical body providing containment and shielding. The body includes a high performance resin developed by TNP.
(ii) A basket providing restraint of the contents.
(iii) An outer frame made of aluminium for handling and storage.
(iv) An anti-vibration system connecting the body to the outer frame. This integrated system is sufficient to meet fuel integrity requirements, thus simplifying the tie-down system of the package during transport.

For Japanese MOX transport, the FS-65 J version, shorter than the FS-65 1300 used for 16 × 16 and 18 × 18 PWR fuel designs, is used for BWR and PWR FAs. The baskets contain either two BWR FAs equipped with fuel holders or one 17 × 17 PWR FA with fuel holder, and fitted in the FS-65 J body. The payload is four FS-65 J packagings per shipment.

In 2000, eight shipments of BWR FAs were made between FBFC International/Dessel and La Hague, where the FAs were transferred into a maritime packaging for transport to Japan.

The packaging types used for sea transport are specially converted spent fuel flasks consisting of TN17/2(M), TN12/2(M) and EXL 4(M) packages. The MOX conversion consists of full decontamination and inspection of the body, then fitting with a MOX basket. A MOX basket containing eight BWR FAs is then fitted into a TN17/2(M) package. The TN12/2(M) can be fitted either with a 12 BWR FA basket or with an 8 PWR FA basket. The EXL 4(M) carries 8 PWR FAs.

(b) For fuels manufactured in Japan

For the delivery of fresh FAs to the Monju FBR, JNC has developed a dedicated transport packaging (Fig. 17).

This packaging consists mainly of an outer shell, neutron shielding material, shock absorbers, a containment vessel and an FA holder. The packaging features advanced technologies in the field of neutron shielding material and an automatic tie-down mechanism for the FAs. Material for the neutron shielding is based on an epoxy resin and has a high hydrogen content compared with that of general resins. An automatic tie-down mechanism for holding the FAs firmly within the packaging is

FIG. 17. Packaging for Monju MOX fresh fuel delivery.

adopted to avoid unnecessary radiation exposure to the operators during the packing operation.

6.3.4. Spent MOX fuel

Spent MOX fuel is carried within TN12/2 B spent fuel packagings. For shielding reasons, four spent MOX FAs are placed in the centre of the basket with eight spent uranium FAs around them. Twenty-five shipments were made to La Hague in 2000.

6.4. ONGOING DEVELOPMENTS

New regulations coming into force require that existing flasks have to be re-licensed or that they need to be replaced by new designs. Some of these new developments are described below.

6.4.1. ANF-18

BNFL are working with Framatome to develop the ANF-18 container for fuel shipments to German power plants. This work combines fuel performance data achieved through package and fuel drop testing with an improved clamping and fuel retention system.

Cogema is considering buying some ANF-18 containers in order to replace the Siemens III type:

(a) As a backup for fuel delivery to the German power plants, pending the availability of the MX6 container as a long term packaging;
(b) For fuel delivery to specific Swiss power plants.

6.4.2. FS-65

BNFL are also collaborating with Transnucléaire to develop the FS-65 (900) package for transports within Europe to specific NPPs in Switzerland. This work involves developing a new concept basket for the FS-65J package which is suited to the specific PWR fuel design.

6.4.3. M4/12

BNFL are themselves developing an advanced MOX transport system called M4/12. This package combines multiple water barrier concepts to enhance criticality safety margins with a patent wall design, which provides improved performance under accident conditions. The concept permits a lightweight design while retaining high capacity.

6.4.4. Advanced MOX casks

BNFL also continue to support Cogema and the Tokyo Electric Power Company (TEPCO) in the development of an advanced MOX cask for marine transport. This development utilizes the conventional heavy flask concept in order to maximize capacity.

Other developments currently being undertaken by Transnucléaire for Cogema include new MOX fresh assembly packagings to replace ageing systems and to add flexibility with respect to modification in fuel designs and plutonium content or isotopic composition [150].

These new packagings are designated MX.

(a) MX1

MX1, which is a development of the FS-65, has been designed in order to meet the need for better contamination monitoring. The upgrade involves suppressing the handling frame of the FS-65 and adding trunnions to the body. MX1 was licensed in France in May 2001.

(b) MX8

MX8 was designed under ST1[2] requirements to replace FS-69 for the delivery of 17 × 17 PWR FAs to EDF plants with the same number of shipments. It allows a higher fissile content and is targeted at the future Framatome/Fragema designs.

---

[2] ST1 is the designation following the IAEA 1996 recommendation before the STR1 appelation.

MX8 is designed for underwater vertical remote unloading and dry vertical remote loading and road transport (Fig. 18).

These main characteristics have led to innovative design work on a 'mid-weight' packaging with original solutions for the body, the basket and the fuel restraining system. Owing to the weight and the road transport requirements, a new high security transport system has been simultaneously developed.

Successful drop tests were performed on prototypes in November 1999; the fabrication of the first packaging ended in November 2000 and thermal tests were run in December 2000. Operational loading tests in the MELOX plant were performed in February 2001 and were followed by unloading tests in real configurations in an EDF NPP in March and April 2001. The licensing certificate is anticipated to be granted in the second quarter of 2001 and the first shipment of MOX FAs is to take place in the second half of 2001.

(c) MX6

The MX6 project was launched at the beginning of 2000 for the European market, focusing firstly on the MOX PWR 16 × 16 and 18 × 18 1300 MW FAs and the MOX BWR 10 × 10 FAs to be delivered in 2003 to German utilities. This design is based on the experience of MX8, but for dry loading and unloading operations.

FIG. 18. The MX8 packaging ready for tests.

The basket concept and the fuel restraining system are specific and developed subject to validation by both Siemens and Framatome/Fragema as fuel vendors. As for MX8, a new transport system is being developed.

The drop tests were performed in April 2001.

6.5. OUTLOOK

Transnucléaire has already demonstrated the feasibility of a higher capacity PuO_2 packaging (the FS-80 concept), although the existing margins of the current FS-47 are sufficient to meet future needs.

BNFL also designed a high capacity PuO_2 package known as the 1680. This package can hold up to 72 kg of plutonium separated from high burnup oxide fuels (Fig. 19). A number of these packages can be carried within a high security vehicle. This package has undergone extensive testing, including high velocity impact testing. Its design allows remote control operations for loading and unloading.

The technical trends concerning fresh MOX shipments are linked to the general trend towards higher burnup, which involves:

- A higher plutonium content, within the authorized limits of MOX plants,
- An increasing percentage of ^{238}Pu due to the higher burnup of the parent uranium fuel.

These higher plutonium contents can affect the economics of MOX transport and lead to greater focus on increased transport capacity per package.

For spent fuel, the input data from the new spent fuel packaging project are consistent with MOX fuel and burnup developments.

More generally, R&D programmes are in place to increase the shielding and criticality efficiencies, which will benefit all future transport packages.

6.6. CONCLUSIONS

Extensive experience has been gained in the transportation of PuO_2 powder, MOX fuel rods and fresh as well as irradiated MOX fuel assemblies. All have been safely transported for more than 35 years, mainly by road but also by sea and air.

The ability to transport MOX safely and economically with rapid turnaround is a central feature of the whole technology of MOX recycling.

Very rigorous transport safety requirements based on the IAEA Regulations for the Safe Transport of Radioactive Material have been adopted by all the countries involved [132]. This means that the packaging systems must be safe under both

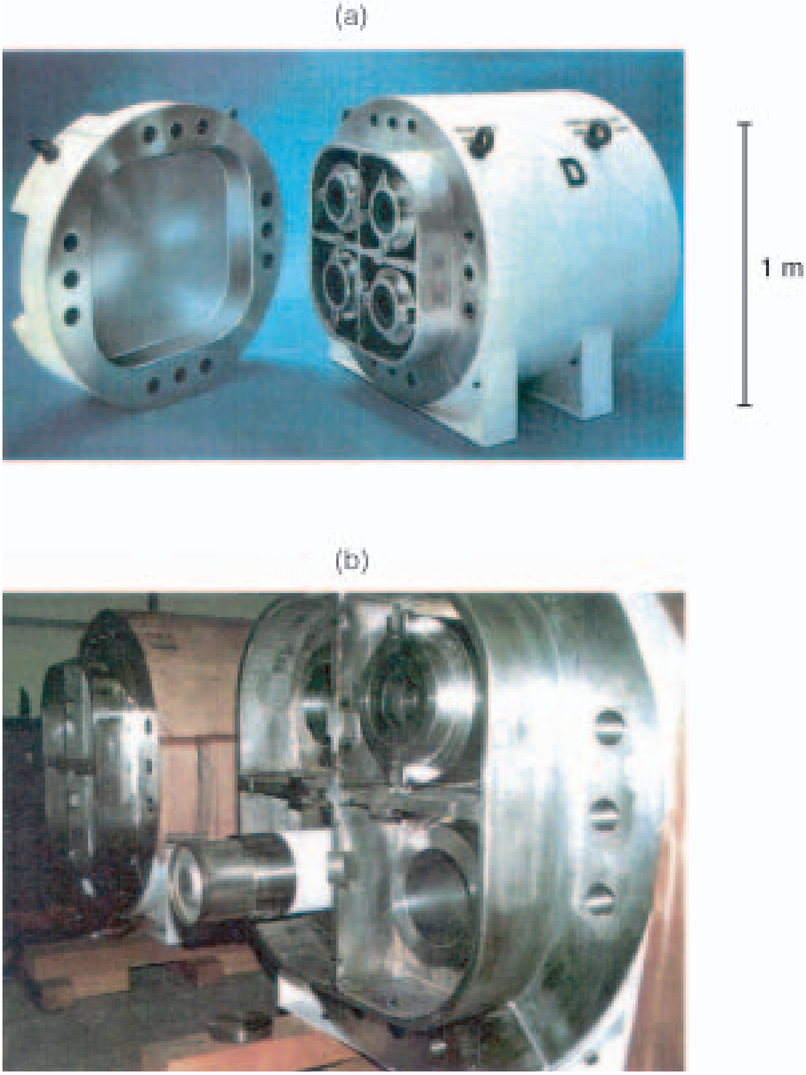

FIG. 19. The 1680 package [151].

normal and accident transport conditions and, additionally, that their designs must account for the fissile nature of their contents.

New packaging designs are being developed to take into account changes in fuel design and composition, and also to reduce transport and licensing costs.

7. SPENT MOX FUEL MANAGEMENT

7.1. SPECIFIC CHARACTERISTICS OF MOX FUELS

The composition of spent MOX fuel, as predicted by computer codes, has larger uncertainty margins than the composition of spent uranium fuel [152]. This is due mainly to the greater complexity of the isotopic chains involved and the smaller experimental database. Broader safety margins may therefore need to be included when examining scenarios and designing equipment and facilities for future larger scale deployment. This situation will improve as post-irradiation examination and related characterization programmes (see, e.g., Ref. [153]) augment the database against which the codes can be benchmarked.

Compared with spent uranium fuel, the fission product spectrum in MOX shows slight differences. Nevertheless, for spent fuel management purposes, these differences have practically no impact on either the radioactivity levels or the radioactivity decay kinetics of the fission products [153]. However, the higher end-of-life reactivity of MOX fuel assemblies results in these fuels operating at higher powers than uranium fuel assemblies during their last reactor cycle.

The higher plutonium contents in spent MOX fuel, especially the higher ^{238}Pu, ^{241}Pu, Am and Cm contents, are by far the largest contributors to making spent fuel management more challenging for MOX than for uranium fuels [154]. The degradation of the plutonium isotopic composition and the buildup of Am and Cm are more pronounced for LWR than for FBR MOX fuels. Therefore, all management steps associated with spent MOX fuel require special precautions to cope with criticality and cooling concerns. Additionally, the high alpha activity (Fig. 20) results in high radiation damage (to which organic materials, for instance, are particularly sensitive) and a high production rate of helium (ultimately increasing pressure in closed confinements). This high alpha activity also translates into a high residual power (decay heat) which depends on discharge burnup (Figs 20 and 21, and Tables XV and XVI). The higher neutron activity, although largely independent of discharge burnup (Fig. 22), also imposes special constraints.

7.2. STORAGE AND FINAL DISPOSAL OPTIONS

The current policy generally adopted for spent MOX fuel disposition is intermediate storage (wet or dry) followed either by reprocessing and recycle or by permanent disposal in a geological repository. The higher residual heat generation in spent MOX fuel (Section 7.1) and the higher residual reactivity require special precautions to be taken in the management of spent MOX fuels compared with

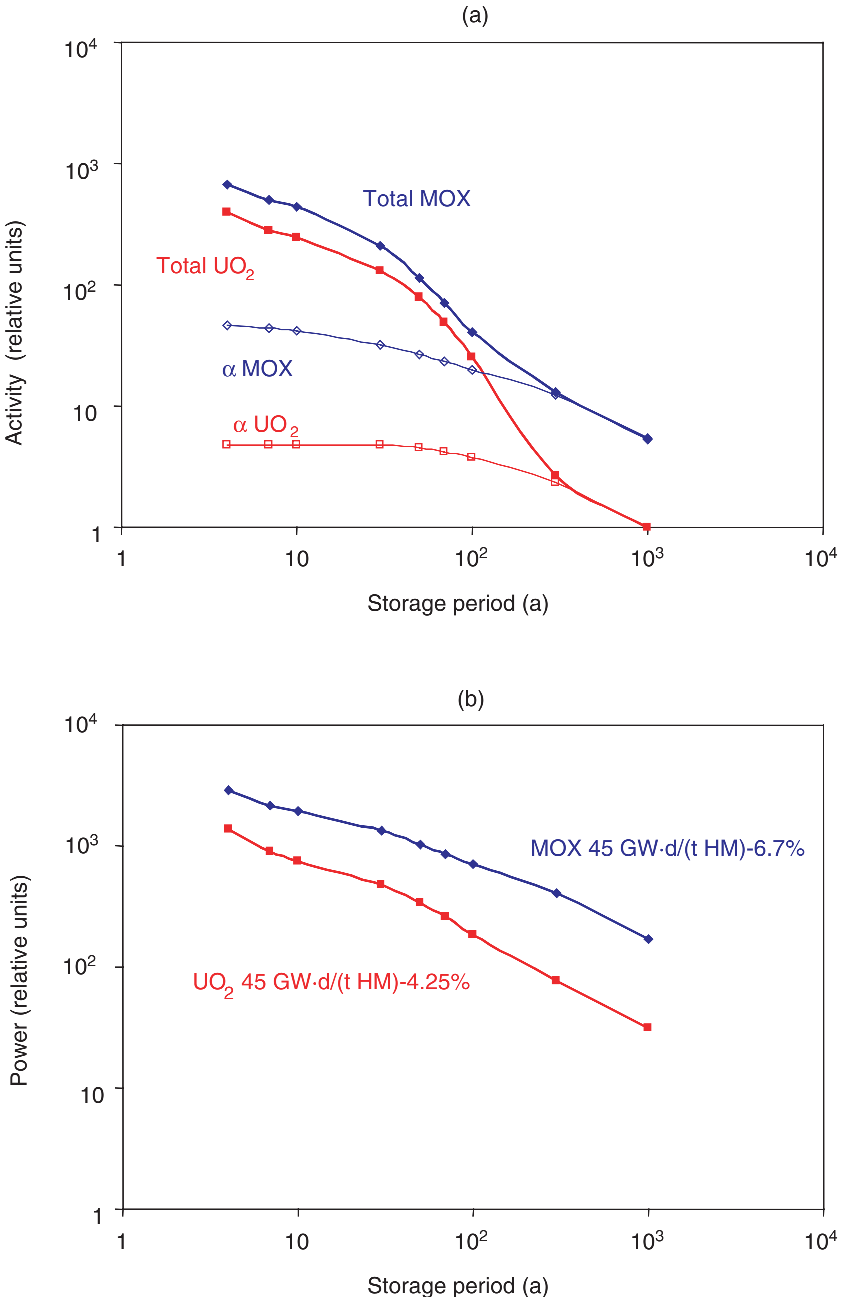

FIG. 20. (a) Total and alpha activities and (b) residual thermal power during storage of UO_2 and MOX assemblies irradiated to 45 GW·d/(t HM) (log–log scales used) [155].

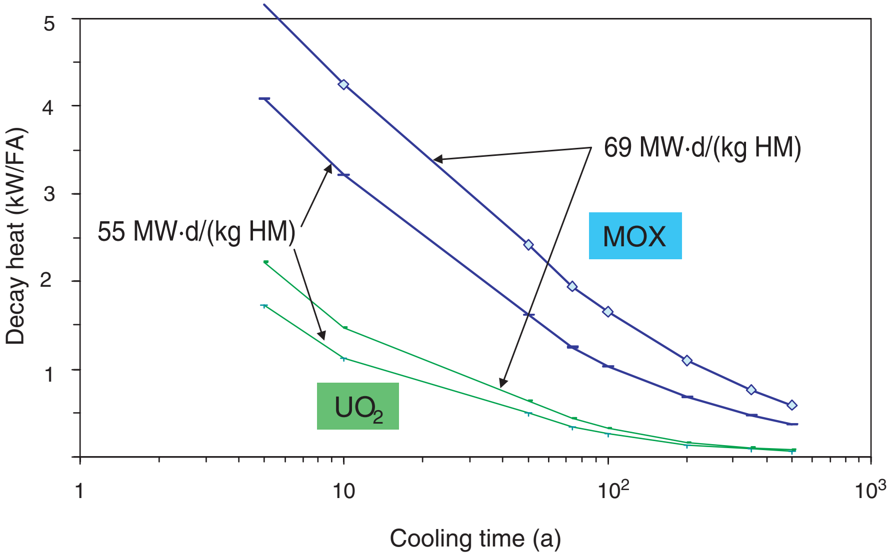

FIG. 21. Decay heat (kW/FA) of UO_2 and MOX FAs as a function of burnup and cooling time (for FA discharge burnups of 55 and 69 GW·d/(t HM)) (logarithmic scale used) [156].

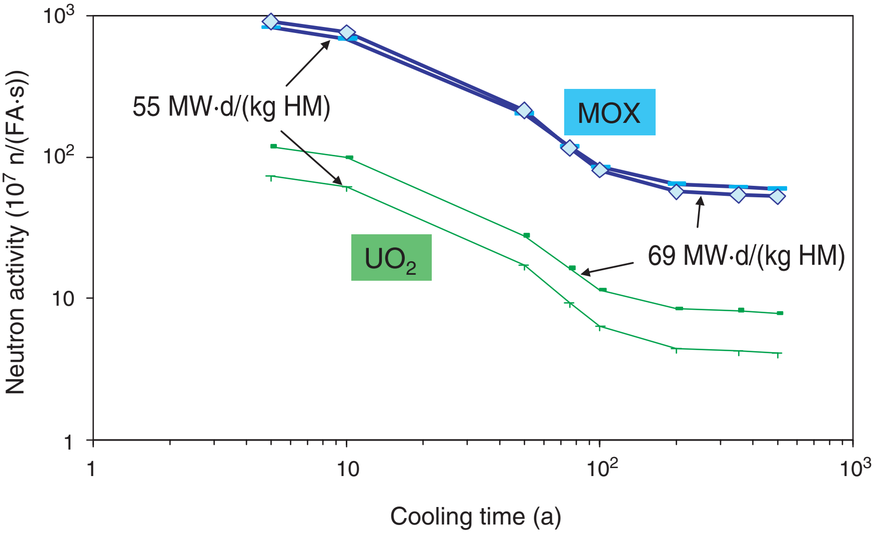

Fig. 22. Neutron activity (10^7 n/(FA·s)) of UO_2 and MOX FAs as a function of burnup and cooling time (for FA discharge burnups of 55 and 69 GW·d/(t HM)) (log–log scales used) [156].

uranium fuels. Impacts are mainly seen in the time delays to be observed before operations such as transportation and disposition.

7.2.1. Interim storage

In wet storage pools, spent MOX fuel assemblies are dispersed among uranium fuel assemblies to mitigate criticality and cooling requirements. This results in no penalty for storage space occupation.

However, the spent MOX fuel assemblies require a longer wet storage period before being loaded into dual purpose dry storage casks. Tables XV and XVI illustrate that the required prior cooling period needs to be two to six times longer for MOX fuel than for uranium fuel, depending on the cask heat transfer limitations. In the casks, the MOX fuel assemblies are also loaded amongst uranium fuel assemblies, meaning that no more storage casks are required than if the reactor had been fuelled only with enriched uranium, provided the discharge burnups are the same (see, e.g., Ref. [157]). The MOX fuel assemblies are positioned at the periphery for heat transfer reasons. The proportion is typically one MOX fuel assembly for three or four uranium fuel assemblies. This does not penalize subsequent transportation for reprocessing if so required, since this is the same relative proportion of assemblies required by the reprocessors (Section 7.3).

7.2.2. Spent fuel disposal

7.2.2.1. Retrievability and reversibility

With the disposal option, a question has to be considered: is retrievability (i.e. the ability to extract the spent fuel from the repository), up to and including

TABLE XV. RESIDUAL POWER (W/kg) OF URANIUM AND MOX FUELS DISCHARGED AT 42 GW·d/t [158]

Cooling time (a)	Uranium fuel[a]		MOX fuel[b]		MOX/U (%)
	Total	Actinides (%)	Total	Actinides (%)	
1	12.3	7	20.5	36	166
2	6.5	7	10.8	45	166
5	2.3	16	4.8	75	206
10	1.5	25	3.6	85	245
50	0.7	47	2.0	93	282
100	0.4	70	1.3	99	344
200	0.2	100	0.9	100	420

[a] Initial enrichment: 3.7%.

[b] Initial Pu/HM: 5.3%.

TABLE XVI. RESIDUAL POWER (W/kg) OF URANIUM AND MOX FUELS DISCHARGED AT 60 GW·d/t [158]

Cooling time (a)	Uranium fuel[a]		MOX fuel[b]		MOX/U (%)
	Total	Actinides (%)	Total	Actinides (%)	
1	16.4	12	28.0	45	170
2	9.3	14	15.7	47	168
5	3.8	27	8.2	68	213
10	2.5	38	6.3	79	255
50	1.1	53	3.1	87	296
100	0.6	72	2.0	94	361
200	0.3	95	1.3	99	451

[a] Initial enrichment: 5%.

[b] Initial Pu/HM: 7.5%.

reversibility (i.e. the ability to modify the spent fuel management option), a requirement or not? If spent fuel is stored indefinitely, the plutonium contained can only be used in the future (e.g., in the form of MOX) if the stored fuel can be retrieved from the repository. Studies in this area have been carried out [159] and are still being performed mainly in Germany [156], as a result of the scheduled shutdown of all NPPs, and in France, as a result of the French law passed in December 1991 on research into radioactive waste management [160, 161].

The drive for reversibility is mainly dictated by flexibility considerations for future reuse of spent fuel components (plutonium in the form of MOX fuel) or future rework (elimination of minor actinides or other long lived radioactive products). Such a view may be justified for several reasons. For example, ethical considerations may prevent us from wishing to impose our present views and concepts on future generations. Scientific and technological improvements may justify implementing partitioning and transmutation of minor actinides or long lived fission products from all spent fuels (even HLW). The economic value of energy resources in spent fuel may increase in the future, justifying the recovery of stored fissile material. Safety, safeguards and/or security regulations may be modified, forcing the retrieval of stored waste. Reasons such as these could justify retrievability; some could justify reversibility.

Currently, almost all countries have opted for retrievability but only France appears to favour spent fuel storage reversibility. The current Commission nationale d'evaluation (CNE) solutions [162] relating to spent fuel storage consider two alternative options:

(1) A deep temporary storage, convertible to a reversible geological storage. The design is identical to one of a permanent repository but with open access points to the underground galleries. The access points will be closed after a sufficiently long period of time has elapsed for reassurance that the repository is fully safe and that reversibility is guaranteed.

(2) A long term surface or subsurface storage, which is much simpler and can be more easily accepted, but which involves the subsequent design and construction of a permanent storage with a reversibility option. Indeed, it is believed that several decades will be necessary to finalize a safe design for the second phase.

To provide full flexibility, one French scenario considers separate storage of spent uranium fuel and spent MOX fuel [163]. In this case, spent MOX fuel has to be stored in the interim above ground facilities for 150 years to reach the same thermal output as spent uranium fuel after 50 years storage. Alternatively, it would need three times as much space in an underground repository if stored for only 50 years. In either option, the cost of spent MOX fuel management is about three times that of spent uranium fuel according to cost estimates provided by the national radwaste agency Andra. Such flexibility would increase the cost of nuclear power generation by about 1%.

7.2.2.2. Plutonium degradation

Current first generation LWR plutonium, i.e. plutonium separated from spent uranium fuels, contains 49–58% ^{239}Pu and 11–15% ^{241}Pu, depending on the reactor type, the discharge burnup and the storage time before reprocessing. However, plutonium in spent MOX fuel from LWRs contains only 41–45% ^{239}Pu and 13–15% ^{241}Pu. After 14 years storage, half of the ^{241}Pu has decayed into americium, increasing the radiotoxicity of the waste and decreasing the fissile worth of the plutonium. This is particularly an issue for re-use in LWRs, where only ^{239}Pu and ^{241}Pu are fissile and, since ^{241}Pu has the higher fissile cross-section, it is the more useful [164]. In FBRs, all the plutonium isotopes are fissile to a certain degree. Thus reprocessing of spent MOX fuel after long term storage or by retrieval from a ‘final’ disposal repository is only credible if FBRs are deployed.

7.2.2.3. Helium buildup

The large buildup of helium pressure during extremely long term storage or final disposal of MOX fuel (Section 7.1) is a topic that has been thoroughly considered [156, 165]. Specific data acquisition programmes are also devoted to this issue (see, e.g., Ref. [133]).

7.3. REPROCESSING OF MOX FUEL

The first experience of MOX reprocessing was with fast reactor fuel. This made use of the same process as that for thermal reactor uranium fuel, but special factors had to be considered to cope with the high plutonium content.

In FBRs, the fuel centre temperature is higher than the oxide pellet sintering temperature (usually 1700°C). As a result, any fabrication heterogeneities are erased during irradiation and the solubility is improved. In addition, the lower production of curium reduces the damage to the tri-butyl phosphate (TBP) and other solvents. Conversely, the higher plutonium contents and the higher fuel burnups constitute more severe parameters for reprocessing.

In LWR MOX fuels, the irradiation temperatures never exceed 1200–1500°C under normal operating conditions and so do not contribute to the ‘erasing’ of heterogeneities of plutonium distribution notable in FBRs. This led to extensive studies on the solubility of LWR MOX fuels.

7.3.1. R&D programmes on MOX fuel reprocessing

7.3.1.1. Laboratory investigations

Reference [7] gives general information of relevance to this section.

The French experience with MOX fuel reprocessing actually began in 1967 with fuels from FBRs (first Rapsodie and then Phénix). The German experience included, in the late 1970s, reprocessing MOX fuel from a PWR (KWO) and a heavy water reactor (HWR), MZFR at the WAK experimental reprocessing facility in Karlsruhe, refabricating the product into MOX fuel at the Alkem plant in Hanau and loading it in the KWO PWR. The Japanese experience began in late 1982 in the Chemical Processing Facility (CPF) devoted to FBR fuel reprocessing.

It was shown (see, e.g., Ref. [166]) that dissolution of the plutonium of those MOX fuels in a nitric medium depended mainly on:

(a) The fabrication of the ceramics: dissolution is very sensitive to the composition of powders and to the heterogeneity of the MOX (UO_2 and PuO_2).

(b) Their irradiation: on the one hand, nuclear reactions consume initial plutonium and generate new plutonium, leading to a more even plutonium distribution as in uranium fuels. On the other hand, the irradiation and the high temperature favour the diffusion of plutonium through the UO_2 matrix; the latter phenomenon differs depending on whether the reactor is an FBR or an LWR.

7.3.1.2. Japanese experiments

Reference [7] gives general information of relevance to this section.

CPF, comprising a series of miniature scale Purex equipment, started operation in late 1982 for laboratory scale hot experiments on the reprocessing of FBR MOX fuels. MOX fuels dedicated to the experiments had reached a burnup of about 100 MW·d/kg in the experimental fast breeder reactors Joyo (Japan), Phénix (France) and the Dounreay Fast Reactor (DFR, United Kingdom). Through experience gained with the different kinds of hot experiments in CPF, recycling of the reprocessed plutonium to the Joyo reactor was achieved in 1984. Results obtained indicated that the fuel dissolution was very satisfactory. The key parameters affecting the rate have been defined. The amount of insoluble residues and their compositions were evaluated by changing the dissolution conditions. This information has been successfully reflected in the design of the continuous dissolver and its operating conditions in the Reprocessing Equipment Testing Facility (RETF). The application of Purex process parameters to FBR fuel reprocessing has also been achieved. For instance, the chemical behaviours of uranium, plutonium, minor actinides and fission products were clarified. Separation of these elements with a sufficient decontamination factor (DF) was obtained in a minimum number of extraction cycles.

To demonstrate multiple recycling of plutonium in ATRs, spent MOX fuel assemblies from Fugen were reprocessed in the Tokai reprocessing plant. The initial fuel assemblies had been fabricated in PFFF during the period 1975–1980 from high quality plutonium (80% Pu_{fiss}). After an average burnup of 10 GW·d/t and two years cooling, the reprocessing took place in 1985.

7.3.1.3. French experiments

Reference [7] gives general information of relevance to this section.

FBR fuel reprocessing in France has been carried out at a number of pilot facilities located at Marcoule and La Hague. The first dedicated plant (Atelier de Retraitement des Combustibles Rapides — Atelier Traitement 1 (AT1)) was sited at La Hague, while the second (Traitement d’oxydes pilote (TOP)) and the third (Atelier pilote de Marcoule (APM)) were sited at Marcoule. These facilities have all been operated by CEA on a pilot scale. A small amount of FBR fuel was also reprocessed at UP2 between 1979 and 1984.

Laboratory scale reprocessing of FBR fuel began in the Cyrano laboratory at Fontenay-aux-Roses in 1968. About 100 kg HM of fuel from the Rapsodie and Phénix reactors were processed there.

The decision to build the AT1 plant at La Hague to reprocess fuel from the Rapsodie fast reactor was made in 1964. AT1 was designed to have the capacity to reprocess one fuel core (130 kg HM) per year and began active operation in 1969, when it processed 220 fuel rods. Rapsodie was fuelled with both plutonium and HEU. In total, some 900 kg HM of Rapsodie plutonium fuel were processed at AT1 by the time it was shut down at the end of 1979.

FBR fuel reprocessing at Marcoule began in 1974 when the Service de l'atelier pilote (SAP), which had previously treated UNGG (similar to Magnox) fuel, was converted to take fuel discharged from Rapsodie. The refurbished TOP plant had a design capacity of 10 kg HM per day, although this appears not to have been achieved. Between 1974 and 1976, fuel discharged from Rapsodie (50 kg HM) was reprocessed. From 1977 to 1983 (when the TOP plant was shut down for refurbishment), only fuel from Phénix was reprocessed there. A total of 9 t HM of Phénix fuel was handled, of which 6.7 t HM was FBR MOX fuel.

In 1979, experiments began with FBR fuel reprocessing at UP2. The fuel was decladed and dissolved in the HAO facility and then sent for chemical separation, diluted in a UNGG fuel solution. A total of 10 t HM of Phénix fuel was treated in this way at UP2 up to 1984.

In 1978, the CEA decided to expand the capacity of the TOP facility. The new APM facility was brought into operation in 1988 with a design throughput of 6 t HM of spent fuel per year. APM served as the head-end for the Traitement d'oxydes rapides (TOR, formerly TOP) separation and materials finishing workshops. Between January 1988 and January 1991, about 5 t HM of fuel from Phénix and the German KNK-II FBRs were processed at APM/TOR. Some LWR MOX fuel has also been processed. Probably about 4 t HM of Phénix fuel were processed, together with the complete first plutonium core of KNK-II.

For LWR MOX fuels, it was necessary to find the degree of heterogeneity that could be tolerated during fabrication for the fuel to dissolve without any particular difficulty in industrial conditions (nitric acid without addition of HF). In this context, studies have been undertaken since 1987 in hot cells at Fontenay-aux-Roses. Solubility measurements were obtained with French and German MOX fuels, before and after irradiation. After irradiation, each dissolution test was performed on the whole rods (1.5–2 kg HM), sheared into 35 mm pieces, in industrial dissolution conditions; before irradiation, the measurements were taken on the pellets. It was experimentally confirmed that the dissolution of MOX fuels was much more complete after irradiation than before. For a burnup limited to 30 GW·d/t, the plutonium solubility of MOX spent fuel proved to be very high, at around 99.77% for all the measurements, whereas, before irradiation, the average solubility was 99.5–99.6%. Processing the residues (dissolution fines) by complementary boiling attack in 10N nitric acid over periods of 6–12 h leads to a recovery of up to 80% of the remaining plutonium in these fines. In practice, the solubility of spent fuel will be even higher, since MOX fuels are currently irradiated to around 45 GW·d/(t HM) and higher.

The dissolution fines from MOX fuels are closely similar to those of uranium fuels. For the same burnup (33 GW·d/t), the total mass of fines ranges from

4.0–4.5 kg/(t HM). Apart from the fact that they may contain PuO_2-rich particles, their composition is roughly similar. The actual mass composition depends on the irradiation history, which is specific to each treated fuel, and, to a large extent, on the dissolution conditions which dictate the solubility of molybdenum, zirconium and tin. Their maximum plutonium content can reach about 0.3%.

In their Marcoule site, the CEA has commissioned a new facility, Atalante, dedicated to R&D on spent fuel processing, with special emphasis on MOX fuels and minor actinides [167].

7.3.1.4. Russian experiments

Reference [168] gives general information of relevance to this section.

Experiments were carried out on samples of high burnup (up to 100 MW·d/kg) MOX fuel from the BR10 and BOR-60 FBRs. High dissolution levels (99.9%) were obtained after redissolution in an HNO_3–HF solution. However, the stainless steel components of the fuel assembly went into solution to an exaggerated extent: 8% in the basic HNO_3 dissolution and 35% in the subsequent HNO_3–HF step. It was concluded that the MOX fuel fabrication needed to be improved for a better U–Pu interdiffusion or solid solution and that a mechanical separation of fuel and structural material would be advisable at the head-end of the reprocessing plant, before dissolution. The extraction chemistry was investigated and it was concluded that a single cycle Purex process was adequate but that it was desirable to consider alternative extraction equipment.

7.3.2. Industrial experience

7.3.2.1. UK experience with fast reactor fuel reprocessing

Reference [7] gives general information of relevance to this section.

Unlike the DFR, PFR was fuelled with MOX. PFR operated for 20 years between reaching criticality in March 1974 and being finally shut down in March 1994. Over that period, as with many demonstration reactors, it operated unevenly, and achieved a cumulative load factor of about 22%. Both core driver fuel and radial blanket fuel were discharged from the reactor, and a total of 93 000 fuel pins were irradiated. This fuel was reprocessed at Dounreay.

The fast reactor reprocessing plant at Dounreay was based upon the earlier Dounreay fast reactor reprocessing facility, building D1206, shut down between 1974 and 1979 for refurbishment. A new fuel disassembly cave, waste facilities and a plutonium production section were commissioned in 1980, and the plant commenced operation in the autumn of that year. By the end of 1993, some 19 t HM of PFR fuel had been reprocessed. Estimating the total weight of driver and blanket fuel irradiated in PFR is complicated because a variety of different fuel pin designs was used to fuel the reactor. Assuming a mean cooling time for the driver fuel of three years, it is estimated that, by the end of 1993, about 20 t HM of driver fuel had been discharged from PFR. In total, the plant has reprocessed over 23 t HM discharged from PFR.

The plant used a modified Purex process (originally developed at Hanford, USA): a three cycle flow sheet with final purification of the plutonium nitrate product. The design of the plant was based on safe-by-geometry mixer settlers.

The dissolver basket contained, typically, 128 pins (i.e. between 4.5 and 5.5 kg plutonium and between 23 and 25 kg uranium). The dissolver cycle time was less than 24 hours, this being the rate limiting operation which constrained the plant throughput to a maximum of around 5 kg plutonium per day for PFR core fuel. The dissolution of irradiated fuel has proved to be efficient and losses of plutonium to insolubles were lower than predicted. Typically more than 99% of the plutonium and 100% of the uranium went into solution in the dissolver. The residual particulate plutonium was removed by a centrifuge. The centrifuge bowls were changed after each dissolution operation and transferred to long term storage as intermediate level waste (ILW). After 1980 the Dounreay plant extracted over 3.5 t Pu. This plutonium nitrate solution has been sent to Sellafield for conversion to oxide.

The fast reactor fuel reprocessing plant was subject to IAEA and Euratom safeguards. Euratom carried out the inspections.

7.3.2.2. French industrial MOX fuel reprocessing

Nearly 28 t HM of FBR MOX fuel have been reprocessed in different French installations (mainly Marcoule and La Hague) without any particular problems and with dissolution yields greater than 99.8–99.9%. This section of the report focuses on the industrial reprocessing of thermal MOX fuel in France [7, 130, 169].

The Marcoule pilot facility (APM, in the south of France) conducted a semi-industrial campaign in early 1992 by reprocessing 2.1 t HM of MOX fuel from the German KKG PWR. Before irradiation, the plutonium solubility was 99.6%. After irradiation to 34 GW·d/t and a cooling time of 3.5 years, the dissolution kinetics observed at APM confirmed the laboratory results mentioned in Section 7.3.1.3. Regarding the material balance of uranium, plutonium and main fission products, consistency was shown between the results of the computer codes and the experimental results. The discrepancies observed in the dissolution liquor for a few fission products (^{106}Ru, ^{125}Sb) can be explained by their poor solubility: these radionuclides are well known components of the dissolution fines. With regard to the observed performance, the residual contamination figure for the hulls was close to the laboratory test values produced during the laboratory investigations (see ‘French experiments’ section above). The dissolution fines recorded after purification contained a small amount of plutonium (about 0.2% of the plutonium present in the fuel). An additional attack test has been performed on these fines, which led to the recovery of less plutonium than expected according to laboratory tests. The dissolution liquor was treated without dilution on the APM extraction line. Plutonium losses in the extraction raffinates and in the unloaded solvent were small. The uranium and plutonium nitrates were concentrated separately. The characteristics of the end products showed that the specifications were met with only two purification cycles.

Following the conclusive experiment with MOX fuel reprocessing at the Marcoule pilot facility, Cogema conducted an industrial reprocessing run in November 1992 with 4.6 t HM of MOX fuel in the UP2-400 plant at La Hague [170, 171]. This fuel, unloaded from the German PWRs KWO, GKN and KKU, was representative of the MOX fuel currently supplied by fuel manufacturing plants, with a plutonium solubility of 99.6–99.8% before irradiation. After irradiation to 33–41 GW·d/(t HM) and a cooling time of five years, the MOX fuel was dissolved in 6N nitric acid (final acidity about 4N) and the average residence time of the hulls in the batch dissolver was four hours. The flow sheets adopted for the purification cycles were similar to those usually applied for uranium fuel. These flow sheets were designed using the process models developed by CEA. The dissolution liquor was diluted in reprocessed uranium dissolution liquor before the first extraction cycle in order to adjust the Pu/U ratio to 2%. The average treatment rate was around 1 t per day in the different facilities. The residual contamination of the hulls, after water rinsing, and the plutonium losses in the purification cycles (extraction raffinate and unloaded solvent) were similar to those usually observed for uranium fuel in the plant. Hence, the higher plutonium flows due to the MOX fuel did not alter the performance. Special analyses were performed on dissolution insolubles separated at the clarification step. The mass of dissolution fines was estimated at about 4 kg/t of uranium and plutonium. Their plutonium content was low (about 0.1% by weight), corresponding to about 0.01% of the plutonium present in the irradiated fuel. All the specifications on uranium and plutonium end products were met.

Another MOX reprocessing campaign on 4.9 t HM of MOX fuel from the Chooz-A PWR (CNA) took place in UP2-400 during the first quarter of 1998. It consisted of 16 fuel assemblies, eight of which were loaded in 1988 and eight in 1989. In the three zoned fuel assemblies, the lowly enriched rods contained 2.7% Pu/HM and the highly enriched rods (constituting the majority) contained 5.8% Pu/HM. The 16 fuel assemblies were unloaded in 1991, when the plant was definitively shut down. As a result, eight assemblies had an average burnup of approximately 27 GW·d/t and the others approximately 18 GW·d/t. The campaign was performed in conditions similar to those employed for uranium fuel reprocessing. It confirmed the previous good results.

For MOX fuel, as for any uranium fuel, the licensing of La Hague is based on a case-by-case authorization by the Direction de la sûreté des installations nucléaires

(DSIN). The reprocessing of soluble MOX fuel is currently authorized in UP2-400. For reasons of economic and technical optimization, Cogema applied for a generic licence to reprocess MOX in UP2-800 and UP3. The requested licence is not unconditional: it rules that, for each fuel type, an additional application will have to be submitted on the basis of a specific safety evaluation. This generic application, amongst others, was submitted for public consultation from February until May 2000. The consultation commission has advised in favour of granting the licence. As at the end of December 2000, the application was still being examined by the competent services within DSIN.

7.3.3. Lessons learned from reprocessing experience

In total, 47 t HM of FBR MOX fuel and 14 t HM of thermal MOX fuel have been reprocessed in Germany, France, Japan and the UK [7].

Experiments conducted in French and Japanese laboratories with MOX fuel showed that irradiation in LWRs led to most of the homogeneity flaws of the plutonium in the tested MOX fuels being erased. The final solubility of plutonium was, consequently, very good (more than 99.97%) and had no effects on the reprocessing conditions, assuming modern fabrication quality MOX fuel.

The feasibility of reprocessing MOX fuel as manufactured today was demonstrated on a semi-industrial scale in APM, and on an industrial scale in the UP2 plant at La Hague, in conditions similar to those employed for reprocessing uranium fuel. The plutonium content of these MOX fuels and their discharge burnup were, however, relatively low compared with those of current and future fuels. It was concluded that the more recent generations of MOX fuel would have to be reprocessed in conjunction with diluting with spent uranium fuel. In practice, MOX fuel must be diluted, when reprocessed, with uranium fuel in a ratio of roughly 1:4. Thus the technological options applied in current industrial reprocessing plants for reprocessing MOX fuel have been confirmed.

The way to multiple recycling is therefore open today and, indeed, plants have already separated second generation plutonium for future use in MOX fuel. However, MOX fuel reprocessing will not be necessary in the next few years, as utilities recycling plutonium are not seeking more material to fabricate into MOX fuel.

7.4. REFABRICATION EXPERIENCE

As mentioned in Section 7.3.1, in the late 1970s, ALKEM refabricated MOX fuel with plutonium recovered by reprocessing spent MOX fuel from KWO and MZFR. This was undiluted second generation plutonium. However, the fairly low burnup of the parent KWO spent fuel and the very low burnup of the parent MZFR spent fuel provided for much less degraded plutonium than what would be considered current pure second generation plutonium. As explained in Section 7.3.3, reprocessing of spent MOX fuel is now conducted by dilution with spent uranium fuel. As a result, the current and future so-called ‘second generation’ plutonium is in fact halfway between first generation plutonium and real second generation plutonium (undiluted). The characteristics of the plutonium processed by ALKEM into MOX fuel are not far from those of the type of plutonium to be expected from current or future reprocessing of spent MOX fuel. During refabrication, ALKEM conducted many characterizations and radiological measurements. The standard process parameters did not need to be adapted and the quality of the fabricated MOX fuel was unaffected. The radioactivity levels at various stages of the fabrication process were much higher than those with first generation plutonium. This was due partly to the higher gamma activity, but predominantly to the higher neutron activity. The refabricated MOX fuel was loaded in the KWO PWR and reached standard discharge burnup without fuel performance or reactor core management problems.

In 1988, the plutonium recovered from spent MOX fuel reprocessing at Tokai (Section 7.3.1.2) was refabricated by PNC into four MOX fuel assemblies, which were loaded in the Fugen ATR. They reached an average burnup of 18 GW·d/t.

Most of the estimated 60 and 140 kg of German plutonium recovered from the reprocessing of MOX spent fuels (Section 7.3.2) were refabricated into fresh MOX fuel at Belgonucleaire/Dessel. No special measures needed to be taken since the quantities were small and the MIMAS process provides for homogenization of the plutonium isotopic compositions.

The estimated 210 kg of French plutonium recovered from the reprocessing of CNA MOX spent fuel in 1998 (Section 7.3.2.2) are still part of the EDF buffer stock and will be refabricated into EDF fuel at MELOX.

Most of the 5.0 t of plutonium resulting from the reprocessing of spent FBR MOX fuel in France has been refabricated into MOX fuel. Some of the 3.7 t of plutonium resulting from the reprocessing of spent FBR MOX fuel in the UK has been refabricated into MOX fuel, but the majority is still stored as PuO_2.

7.5. FUTURE TRENDS

Owing to the time delay to be observed before reprocessing, the valuable ^{241}Pu decays into ^{241}Am. This adds further to the high Am and Cm contents of MOX spent fuel. The disposition of these two elements has to be, and is being, considered in long term strategy definitions. In this context, alternative reprocessing technologies are being investigated which would permit reprocessing to be performed after a shorter cooling time. This would avoid losing ^{241}Pu and would reduce the ^{241}Am inventory to be managed.

7.6. CONCLUSIONS

Spent MOX fuel differs from spent uranium fuel principally in having a slower decrease of decay heat generation, a larger fissile inventory and an increased content of minor actinides (americium and curium). All these have to be taken into account for spent nuclear fuel management but they do not prevent the storage and disposition technologies established for uranium fuels being applied to MOX fuels. Both the direct disposal and the reprocessing options can be envisaged as being applicable for spent MOX fuel. Reprocessing, however, benefits from an already well established database. The refabrication of plutonium separated from spent MOX fuel has been carried out in MOX fuel fabrication plants.

8. WASTE TREATMENT AND DECOMMISSIONING

8.1. INTRODUCTION

Decommissioning is the series of actions that are taken to allow the removal of some or all of the regulatory controls that have been placed on a facility that has used radioactive material. These actions include both administrative and technical actions, and may range from dismantling an entire system or building(s) to performing some decontamination and a radiological survey to show that acceptable conditions have been met.

The topic of decommissioning also includes the waste treatment of arisings from the decommissioning process. The significant aims and policies behind decommissioning include:

- minimization of volumes that have to be disposed of as radioactive waste
- recycle of material and facilities where possible
- release of material from regulatory control
- protection of human health and safety
- environmental considerations.

Decommissioning of plutonium facilities follows similar rules to other nuclear facilities. References [172, 173] provide a comprehensive summary of decommissioning requirements, techniques and experience. The nature of plutonium facilities, however, is such that the presence of plutonium leads to special considerations when planning decommissioning activities. The measurement of the residual inventory to avoid criticality hazards and strict control of containment and ventilation are significant considerations in the decommissioning process.

This section discusses the historical background to, and strategy for, the decommissioning of plutonium active facilities, looking at some of the problems encountered and solved and how these may apply to future decommissioning projects. A number of facilities that have been, are in the process of being or will in the future be decommissioned are described, and the future perspectives of the industry in this regard are reviewed. The discussion includes decommissioning from an early stage in plant design (which now occurs regularly) and using lessons learnt from previous experience. Note that decommissioning of non-plutonium nuclear facilities is not addressed in this section but information on such facilities can be found in Ref. [172].

8.2. BACKGROUND

8.2.1. Issues in decommissioning plutonium facilities

Plutonium active fabrication facilities have been used for a number of functions, from plutonium metal production for defence related purposes to plutonium oxide production for storage or use in fuels. While commercial scale MOX fuel plants are now in use, older facilities were often on a pilot plant or laboratory scale. Usually relatively small, particularly compared with reprocessing plants, they are generally based on a building(s) housing gloveboxes, ventilation ducts, filter banks, services and associated equipment.

Major considerations in the decommissioning of MOX and plutonium production facilities arise from the presence of plutonium. From a dismantling point of view, PuO_2 and $(U, Pu)O_2$ will exist in some parts of the process. The main considerations, therefore, for the safe decommissioning of such a facility are [172, 173]:

- Measurement of residual inventory to avoid criticality hazards;
- Strict control of containment and ventilation to restrict contamination;
- Accurate assay of waste arisings (decontamination and segregation as necessary);
- Possibility, and hence mitigation, of significant operator radiation exposure;
- Management of waste and other products.

The need for alpha protection during dismantling is a particular issue with these facilities [172]. An important consequence of higher surface contamination levels and the presence of alpha emitters (e.g. where contaminated plutonium gloveboxes have to be dismantled) is that particular attention needs to be paid to personnel protection against the greater potential for internal exposure from inhalation or ingestion.

Reprocessing plants (the decommissioning of which is not discussed in detail in this section) not only encompass alpha active areas to be decommissioned but also offer slightly different challenges. The principal two challenges are the high radioactivity levels inside certain cells due to fission products and the presence of all types of contamination (alpha, beta and gamma emitting radionuclides). The possibilities of changing locations and concentrations of fissile materials and alpha emitters in confined spaces, and consequential difficulties with detection, are also major considerations. The primary radiological concerns also change with time when dealing with alpha active plants, because of the decay of ^{241}Pu to ^{241}Am.

The features of alpha active facilities vary widely and, as such, so do decommissioning requirements.

8.2.2. Strategy and regulatory controls on decommissioning activities

The strategy and planning for decommissioning nuclear facilities are now key aspects in the nuclear programme and routinely use the as low as reasonably practicable (ALARP) principle.

Reasons for decommissioning a nuclear facility include:

- change in government policy
- obsolete technology
- uneconomic operation of the facility
- unplanned event or accident
- safety issues
- programme completion
- desire to reuse the building.

Decommissioning programmes seek to achieve minimum discounted overall costs, recognized plant risk, surveillance and maintenance costs, waste route availability and interaction with other plants. Plutonium plants have a higher priority for early decommissioning than, for example, standard reactors, because they are prone to deterioration of plant and equipment due to the high activity and the increase of radiation levels due to americium in-growth. The principles of risk management are heavily utilized in the process of decommissioning.

Once decommissioning has been decided on, three main options are available either alone or in combination:

(1) Immediate dismantling of the facility,

(2) Safe storage of the facility,
(3) Deferred dismantling and eventual entombment of the facility.

Deferral of decommissioning activities may not significantly reduce the activity of the remaining radionuclides, the quantities of radioactive waste produced or the radiation exposure of site personnel [173]. This is due to the relatively long half-lives of the radionuclides involved and, in some cases (such as americium), the potential for isotopic in-growth. Additional disadvantages in delaying the decommissioning activities include the gradual deterioration of structures, systems and components designed to act as barriers between the inventory of radionuclides and the environment, and the loss of operating personnel experience. The deterioration may also apply to systems that could be used during dismantling of the facility.

Currently, nuclear licensing arrangements need to include requirements for the licensee to prepare decommissioning plans and programmes [173]. Decommissioning projects have an overall safety case as an initial requirement and also provide detailed phase submissions at the appropriate time. Particular attention is paid to the projected dose uptake in the safety justification stage, via the safety case and formalized ALARP dose studies.

8.3. WASTE TREATMENT OF MOX ARISINGS

Various types of transuranic waste (TRU) are produced in the course of MOX fuel fabrication and plant decommissioning. Organic and inorganic solid wastes arise in gloveboxes during maintenance and repair and when working equipment is retired. Liquid waste is generated during chemical process steps as filtrates from plutonium conversion and analytical procedures in laboratories.

All wastes generated in the controlled areas outside the gloveboxes (room waste) are to be handled as active waste. To reduce the volume of this room waste, the organic part is often incinerated. The ashes produced and the inorganic part of the room waste can be compacted to pellets, which are stored in 200 L drums as an interim waste product.

There is a requirement to minimize waste throughout the fuel cycle. Categories of waste are generally dealt with in a similar manner whether they arise from fabrication, decommissioning or other operations. Fabrication plants aim to use as much waste as possible in recycle, for instance. The minimization of plutonium in waste streams in particular is desirable because the presence of plutonium requires special care in handling and disposal. Off-gases and ventilation gases are filtered and released. All secondary waste is collected, stored, treated, conditioned and packaged as required for storage in dedicated facilities prior to disposal. The treatment of this waste is performed using:

(a) For solid wastes: Incinerators and compactors;
(b) For liquid wastes: Evaporators, ion exchange columns, precipitation tanks, cementation units, storage tanks and associated pipework.

Sections 8.3.1 and 8.3.2 give examples of the waste treatment of MOX arisings from Europe and Japan, respectively. These discussions include waste arisings from MOX fabrication processes; they are not limited to decommissioning activities.

8.3.1. European experience of MOX waste treatment

Waste arisings originating from plant operation and maintenance, as well as waste due to originate from plant backfitting and ultimately decommissioning, have also received proper attention. Indeed, waste management influences fabrication costs, personnel exposure, licensability and public acceptance. At the Belgonucleaire plant, by identifying and optimizing the waste generating operations and by educating the personnel, plutonium contained in the waste has been reduced to less than 0.1% of the plutonium contained in the delivered MOX fuel [174], notwithstanding the policy of not stripping plutonium from the waste. This illustrates the accomplishments achievable by feedback from lessons learned. At MELOX, licensing authorization is imposed to reduce the radioactivity releases and the plutonium wastage to almost zero. Dedicated facilities have been commissioned at the MELOX/Marcoule site and at La Hague to achieve this target [175–177]. In addition to this effort of all the manufacturing plants to minimize plutonium in the waste streams, improvement programmes are also being pursued to reduce the volumes of each radioactive waste category [178].

8.3.2. Japanese experience of MOX waste treatment

Currently, transuranium waste, generated from MOX fuel fabrication by JNC, is classified into combustible, chlorine containing or incombustible waste when it is taken out from the process and put into sealed plastic packages. These classified wastes are placed in 200 L drums or metal cubic containers for storing at a storage facility. Incinerators dedicated for the first two categories are utilized for the purpose of reduction and stabilization treatment as a pre-conditioning step for the further development of technology to melt and solidify the ash of incinerated wastes by microwave heating (which is currently in the demonstration phase). To reduce volume and decontaminate incombustible waste by creating metal ingots, electro-slag re-melting techniques are under development. This method uses tungsten electrodes to melt small pieces of metal waste by the Joule heat generated from pre-fed molten slag of oxide material mainly composed of Al_2O_3 and CaO.

8.4. DECOMMISSIONING TECHNIQUES

Decommissioning techniques fall into the following broad categories:

(a) *Measurement and characterization,* which includes the assay and assessment of the radiological contents of the facility and provides valuable information for the planning of the overall decommissioning project.

(b) *Decontamination of systems and dismantled components* may be performed depending on the balance of disposal costs of materials with and without decontamination. There are numerous methods available for use [179].

(c) *Dismantling, segmenting and packaging* techniques may be limited or determined by the working conditions. For instance, plasma cutting is preferable to grinding of highly contaminated material because of the amount of airborne contamination generated. Figure 23 shows an example of a remote

FIG. 23. Example of a remote handling machine.

cutting technique that was used in the decommissioning of the MOX fabrication facility at Sellafield (Section 8.5.5).

(d) *Handling, remote handling and handling waste.* There are arguments for minimizing manual handling and utilizing remote techniques in all areas of decommissioning, although issues such as speed of operation and intervention for maintenance may offset those arguments.

(e) *Decontamination of buildings and structures.* Decontamination is a major decommissioning activity and, as a minimum in decommissioning projects, the floor, walls and external structure surfaces are decontaminated by surface cleaning or surface removal [179].

(f) *Protection techniques.*

8.5. EXAMPLES OF DECOMMISSIONING ACTIVITIES

The examples described below cover a range of facilities in a range of countries. The technical lessons learned in decommissioning are to some extent shared across the world and some companies are developing significant experience and expertise in this area.

8.5.1. French experience

Decommissioning of several plutonium contaminated facilities has been carried out in France. This has mainly addressed laboratories devoted to the early development of FBR fuels and facilities connected with civil and military plutonium.

An example of this is the decommissioning of the Fontenay-aux-Rose laboratories, which became necessary owing to the expansion of the urban areas surrounding Paris. Another example is the UP1 reprocessing facilities at the Cogema/Marcoule site. These shut down permanently in 1997 after 40 years of operation during which over 18 000 t of GCR fuel was reprocessed to separate the plutonium required for the French programme. The dismantling and site clean-out project is expected to take more than 30 years including final shutdown (incorporating rinsing and decontamination), decommissioning itself and retrieval and conditioning of all the waste generated during the operation of the plant. The dismantling operations started in 2000 and will last until 2015.

8.5.2. German experience

The ‘Plutonium-Testextraktionsanlage’ (Pute), which started operation in 1980, was a complete extraction cycle of the Purex process and, at the time, was considered the largest glovebox facility for pulsed sieve plate columns in the world.

It was operated until 1991 as a test facility for the optimization of plutonium purification. With a fixed inventory of 4 kg, a total quantity of 390 kg of plutonium was processed in its lifetime without incident. The money spent on dismantling the facility was about 45% of the construction costs. Dismantling took about two years and was completed in 1996 [180]. The remaining structures, essentially an empty steel caisson with air inlet and outlet lines, have been decontaminated. Future surveillance requirements are dependent on the results of measurements of its surface condition.

The Siemens MOX Facility, Hanau, formerly ALKEM, which had processed 8.5 t plutonium into fuel for fast and thermal reactors since 1970, was shut down in mid-1991 after a minor incident in its storage area, owing to the anti-nuclear policy of the local government. In 1995, Siemens and the German nuclear utilities decided to abandon MOX fabrication in Germany, leading to a programme to decommission the facility [181–185].

At that time, roughly 1 t of plutonium was being stored as powder and solutions, i.e. in a state deemed not suitable for long term storage or transport. Thus, in 1997, clean-out of the plant was started and by 2001 all those materials had been processed into rods and assemblies (very similar to standard fuel rods and fuel assemblies), either for long term storage or for transport to a reprocessing facility.

In 2000 the decommissioning of the facility itself began — this is scheduled to be finished in 2004. The decommissioning makes use of the vast experience gained during 30 years of maintenance, repair and replacement work connected with the operation of a MOX facility. The main steps can be characterized as follows:

(a) Removal of working stock and hold-up from gloveboxes for the purpose of dose rate reduction.
(b) Measurement of plutonium in gloveboxes (using a newly developed system with a detection limit of 1g of plutonium).
(c) For standard size gloveboxes, insertion into a special dismantling glovebox, and cutting and depositing of pieces into waste drums (200 L standard).
(d) For large size glovebox compounds, construction of a plastic tent all around, and cutting and depositing of pieces into 200 L waste drums.
(e) Contamination check of floors, walls and building structures, removal of contaminated materials to 200 L waste drums.
(f) Removal of floors and the complete building structure for unrestricted disposal or reuse.
(g) Measurement of plutonium content in drums (newly developed system) so as to comply with German disposal regulations.
(h) Void filling of waste drums with cement.
(i) Embedding of drums into large cuboid sized steel containers, suitable for final disposal.

(j) Backfilling with concrete.
(k) Storage in intermediate storage building, awaiting final disposal. Containers are qualified for at least 40 years interim storage.

German regulations governing final storage of TRU waste stipulate that all waste types be solidified and embedded in a solid matrix. On the basis of its extensive experience in this field, Siemens therefore decided in the early 1980s to develop a cementation process that allowed the various waste types to be safely and reliably immobilized in a single waste product in a one step process. These operations are generally carried out at atmospheric pressure and ambient temperature and therefore pose no fire hazard. Figure 24 presents an overview of the waste generation, treatment and conditioning process.

The drums used for the cementation product were designed for a storage period of approximately 20 years. The German government now estimates that a geological repository will not be needed for at least another 30 years. The present goal is therefore to place the waste drums in an additional package suitable for long term interim storage that needs no maintenance and will probably not require any further conditioning of the waste package prior to final storage.

According to the waste acceptance criteria specified for the Konrad repository being planned in Germany, and with a view to transport management, waste shall not be stored in single drums. Planning foresees the use of only large volume cuboid

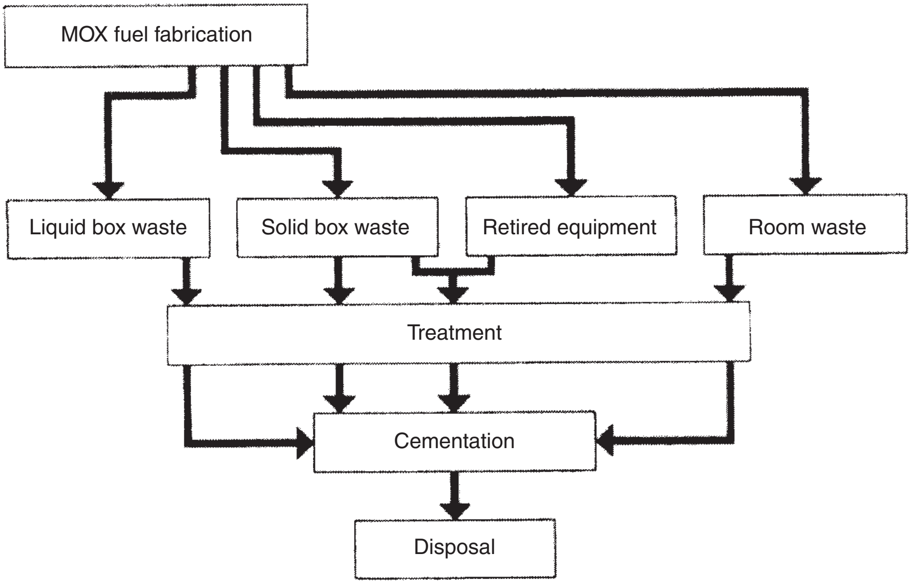

FIG. 24. Scheme for treatment of TRU waste arising from the fabrication of MOX fuel.

shaped ‘Konrad containers’. For example, ten drums can be stored in a single Konrad container with 5.4 m^3 disposal volume. The voids between the drums will be filled with concrete, thus achieving a final product stable against pressure acting from outside the container.

Descriptions of other decommissioning experience in Germany can be found in Refs [186–190].

8.5.3. Italian experience

Decommissioning of a plutonium fabrication plant at the Casaccia Centre (operated from 1971 to 1987 in manufacturing fuel for the Garigliano BWR, FBR programme and CANDU reactor) consisted of dismantling 60 gloveboxes and the equipment contained within them, and cutting and packaging them into 200 L drums. Dismantling took place in a carbon steel containment box, whose internal surface was protected by a strippable coating. Workers in ventilated suits performed the size reduction within the box with mechanical tools. The plutonium contaminated waste was monitored by non-destructive systems based on gamma and passive neutron assay [191].

8.5.4. Japanese experience

Decommissioning of MOX fuel facilities has not yet started in Japan. However, JNC has accumulated experience of dismantling MOX gloveboxes when they became surplus to requirements in its plants. The main experience from PFFF, the older facility, includes withdrawals of the co-conversion test line, wet recovery process and fuel pin/assembly line of the former Joyo fuel fabrication plant. Decontamination and segmentation of equipment were carried out by manual operation prior to dismantling the glovebox. Greenhouse structures and ventilated suits for the purpose of radiation protection were utilized during the dismantling operation. In total over 450 m^3 of glovebox have been dismantled in PFFF.

On the basis of the experience gained with PFFF noted above, a dismantling system incorporating a large glovebox has been installed in the PFPF — this glovebox can accommodate any process glovebox requiring to be dismantled. The PFPF process gloveboxes have a unique feature incorporating a double door connection with the material transfer system, which allows them to be detached from the process line independently when needed. With these features, PFPF has enabled glovebox dismantling to take place while the plant is in operation. The system is also equipped with an assay system to measure the plutonium contained in 200 L waste drums or rectangular containers for material accountancy and safeguards purposes. So far, eight gloveboxes totalling 72 m^3 in volume have been dismantled using this system.

8.5.5. British experience

BNFL has experience of decommissioning MOX fabrication facilities. The PFR fuel fabrication facility operated at Sellafield from 1974 to 1988. The co-precipitation and dry granulation plants associated with the conversion of recovered uranium and plutonium for the PFR programme have already been totally decommissioned. Operations on these plants allowed the testing of improvements in techniques and equipment for temporary containment, contamination control and decontamination methods, for instance, while also developing in situ plutonium assay, recovery and size reduction methods. Typical operations included removal of large items from gloveboxes, in situ size reduction of boxes and plant items and disconnection without loss of containment. This project was hampered, as was the case with a number of older plants, by inadequate clean-out at the end of operations and inadequate design information on plant modifications, for instance, all resulting in extra dose uptake and greater operational restrictions during decommissioning.

The PFR fabrication facility itself is partially decommissioned. Decommissioning of the final assembly and pin filling lines, including the vibro-compaction area, was completed from 1993 to 1996. Figure 25 shows the active canning line prior to decommissioning. The result of the decommissioning of the final assembly and pin preparation area is illustrated in Fig. 26. The area is now operating as a plutonium contaminated material (PCM) drum store. The final phase of decommissioning the remainder of the fuel line is about to begin and concerns the most heavily contaminated, highest dose region of the facility where pellet preparation occurred. Remote handling techniques are to be used [192].

The UKAEA have also dealt with decommissioning of MOX and advanced fuels fabrication facilities. Similar techniques to those used in previous projects were utilized, using hands-on dismantling inside a modular containment to prevent spread of contamination. Airborne contamination was controlled by the flow of air through a filtered extraction system [193].

8.5.6. US experience

Several MOX or plutonium contaminated facilities have been decommissioned in the US, some of them to a level which would allow unrestricted use of the site [172]. Others were decontaminated, dismantled and put into either an active or passive safe enclosure condition. A policy of waste minimization is followed with all decommissioning programmes.

A plutonium processing facility at the Savannah River site in North Carolina has been undergoing decommissioning since 1984 [194]. From receipt of acid solutions from reprocessing, oxalates were precipitated and calcined into an oxide form. The plant also processed ^{237}Np, ^{239}Pu, ^{233}U and ^{235}U as well as ^{238}Pu. As a

FIG. 25. The active canning line of the PFR fuel fabrication plant at Sellafield (before decommissioning).

FIG. 26. The PCM drum store, originally the final assembly and pin preparation area of the PFR fuel fabrication plant at Sellafield (after decommissioning).

result, the process areas were highly contaminated. All lines were flushed, cabinets emptied and interiors decontaminated and painted to fix residual contamination.

A number of MOX fabrication facilities have been decommissioned in the USA. The Advanced Fuels Laboratory at Vallecitos, California was used from 1962 to 1979 for MOX fuel fabrication and development, including FBR fuel development. Plutonium was first removed from the site, followed by cleaning of the process equipment, size reduction, removal and packaging for disposal. Glovebox interiors were then painted to fix residual contamination. The MOX fuel fabrication plant at Cimarron, Oklahoma, was also decontaminated to levels that allowed the reuse of significant amounts of plant and equipment. Planning started in 1987 at another MOX fuel fabrication facility in Tennessee. Stainless steel containment has been constructed to house the sectioning and decontamination works. Decontamination work is to be carried out by means of an ultrahigh pressure water jetting system incorporating a recirculated medium and by material control accountability using a non-destructive neutron assay system.

8.5.7. The Eurochemic finishing line

The Eurochemic reprocessing plant [172] at the Dessel site was owned by a 13 nation consortium and operated between 1966 and 1975. After shutdown, the reprocessing plant was decontaminated from 1975 to 1979 to maintain a safe standby condition at a reasonable cost. Belgoprocess was established to take charge of activities on the site and became a subsidiary of NIRAS/ONDRAF at the end of 1986.

The aim of decommissioning the Eurochemic reprocessing plant was principally to maintain a safe standby condition at a reasonable cost, which was achieved by 1979. From 1987 to 1990 two storage buildings for uranyl nitrate, plutonium dioxide and spent solvents were emptied and decontaminated as an exercise to prove the feasibility of restoring greenfield conditions. Checking techniques and costs and training of personnel was an aim of this work prior to dismantling the major part of the Eurochemic reprocessing plant. The main building had seven floor levels and contained about 1150 t of metal. Decommissioning operations in this building are now being carried out on an industrial scale and are planned to last until 2003.

Metal components are dismantled by plasma cutting. Horizontal plane movements are remotely operated as they are mounted on a rail system. Limited diameter pipes are cut with hydraulic shears. Dry cutting of cast iron shielding blocks and concrete blocks is carried out using hydraulically controlled saws with cutting segments of a mixture of polycrystalline diamond and tungsten carbide. A floor shaving machine was introduced to decontaminate concrete surfaces where possible.

Recently, a comparative demonstration programme on dry and wet abrasive blasting techniques, to decontaminate metallic components, was completed. This showed that it is economically viable to decontaminate such components to clearance levels when all costs for conditioning and disposal of the resulting wastes are considered.

8.5.8. Weapons plutonium facilities

The issue of weapons grade plutonium is of huge significance in the area of decommissioning. Several plutonium contaminated facilities are legacies of nuclear weapons development facilities. This is a bigger problem in the Russian Federation and the USA than in other countries. Significant funds and resources are required to address this problem.

The very high ^{239}Pu content of weapons grade plutonium simplifies handling and processing (as there is less gamma radiation), but criticality issues mean precautions are necessary, leading to adaptation of the equipment required.

8.6. FUTURE PERSPECTIVES

A number of countries have already completed decommissioning projects on plutonium active facilities as discussed above. There are, however, significant projects to come in the future particularly in, for example, France and Japan, both of which have a strong interest in MOX but which have yet to make significant progress in decommissioning their alpha active facilities. These projects will benefit from the experience of past facilities and also from early consideration of decommissioning at the design process stage [173].

The requirements and costs of decommissioning are now much better understood across the nuclear industry and the importance of including decommissioning considerations in nuclear facility design is recognized — indeed there are regulatory and licensing requirements on this issue. The objectives are to reduce costs, financial and otherwise, by reducing worker exposure, minimizing waste generation and simplifying dismantling procedures. This planning for the future also provides assurance about the environmental impact of all aspects of nuclear power for the public. A further advantage of designing for decommissioning is the ability to use remote techniques for standard decommissioning operations, which may not have been possible if consideration had not been given prior to construction. Remote operation becomes less practicable if the facility is not designed to be dismantled.

Plants, such as SMP, which are undergoing active commissioning, incorporated significant decommissioning experience into their design. The initial planning aims to reduce the complexity and cost of the decommissioning procedure when it becomes necessary at the end of plant life at some time in the future. PFPF has also taken the need for dismantling into consideration in its plant design as described in Section 8.5.4 above and the same is also expected to be true with the Japanese commercial MOX plant, construction on which is due to start in 2004.

Dose is usually the significant factor in deciding between remote and manual options, although there is generally a significant financial cost increase with the introduction of remote techniques to be borne in mind. Problems with dose levels have been incurred previously due to the length of time the facility was left prior to decommissioning. Future plants are likely to be decommissioned as soon as practicable after the end of operations [173]. Plant operators will immediately perform a post-operational clean-out (POCO). Alpha facilities will then be dealt with immediately, as opposed to the strategy with beta–gamma facilities, where decommissioning commences after a period of time to allow for decay of certain isotopes.

Simplistically, decommissioning implies size reduction of materials to 200 L PCM drums (or equivalent) and movement to a storage facility. The longer term issue of final waste treatment and storage has yet to be resolved and storage facilities await a final solution. There may be international agreement in the future over suitable storage strategies.

8.7. CONCLUSIONS

Plutonium plants have a higher priority for early decommissioning than, for example, standard facilities because they are prone to deterioration of plant and equipment due to the high activity and the increase of radiation levels resulting from americium in-growth. The measurement of residual inventory to avoid criticality hazards and strict control of containment and ventilation are major considerations in the decommissioning process.

A number of significant decommissioning projects will be undertaken in the coming years. These must learn from the experience of past facilities and should also take note of the benefits to be drawn from early consideration of decommissioning at the process design stage. The benefits of incorporating remote techniques for standard decommissioning operations should not be underestimated.

The drive to minimize plutonium levels in waste streams and reduce the volumes of waste themselves will continue. Current policies of optimizing waste generating operations and educating personnel, driven by licensing authorities and the need to address public concerns, has to be maintained.

9. APPLICATION OF SAFEGUARDS AND PHYSICAL PROTECTION TO MOX FUEL

9.1. INTRODUCTION

One of the principal concerns associated with expanding nuclear power applications is the possibility that a State might attempt to acquire fissile material for use in nuclear weapons or other nuclear explosive devices under the guise of a 'peaceful' nuclear power programme. All isotopes of plutonium have fission properties suitable for use in nuclear explosives [195], but ^{239}Pu is ideal for weapons use and thus States that developed nuclear weapons chose either highly enriched uranium or plutonium with 93% or more ^{239}Pu content. Indeed, heat emission, spontaneous fission neutrons and gamma radiation pose increasing engineering problems as the content of ^{239}Pu decreases in relation to the other isotopes. Nuclear explosives manufactured from suboptimal isotopic composition fissile material cannot be as efficient or as sophisticated as nuclear weapons need to be. They are not as stable, have arbitrary yield, are generally not tested and each such explosive may be even 'one of a kind'. Nevertheless, the international community has agreed, and the IAEA has ruled, that all plutonium (except for 'heat source' plutonium containing 80% or more ^{238}Pu, for which the engineering problems associated with its use are considered too great to warrant credible interest) must be considered on an equal basis from the safeguards and physical protection points of view.

Despite the proliferation concerns, the development of a closed fuel cycle including the use of MOX fuel started almost simultaneously with the deployment of nuclear electricity generation. The safeguarding and physical protection of plutonium in large bulk handling facilities are therefore not new concepts. The necessary approaches and techniques have been progressively developed for reprocessing plants, MOX fabrication plants and reactors using MOX fuel over about thirty years in Europe and in Japan. Close co-operation between the IAEA, the 'State's system of accounting for and control of nuclear material' (SSAC) and Euratom for regional safeguards systems, as well as with the facility operators themselves, ensures that duplication of inspections is prevented. In particular, the New Partnership Approach between the IAEA and the Euratom Safeguards Office has allowed the IAEA to reduce significantly its resources spent in the European Union, which is where most civil plutonium is currently separated, refabricated into MOX fuel and utilized in NPPs (Sections 3–6). Although recognizing the important part played by Euratom as a regional safeguards organization [196], this section mainly focuses on the implementation experiences of the IAEA as the responsible organization within the framework of the Non-Proliferation Treaty (NPT).

It should be mentioned that international safeguards apply only to the IAEA Member States that have signed the NPT and only to plutonium declared to be for peaceful uses. Defence secrecy reasons have prevented the NPT having generic application to military plutonium (and uranium). This inherent discrimination in the NPT has made it unacceptable to some Member States and has reduced its field of application in other Member States [197]. It should be noted, however, that under new verification agreements planned between the IAEA, the USA and the Russian Federation, these States intend to submit their weapons origin fissile material to IAEA verification [198].

As mentioned above, in applying safeguards at reprocessing plants, MOX fuel fabrication facilities and MOX fuelled reactors, the IAEA bases its requirements on the assumption that plutonium with any combination of the isotopes encountered in nuclear power activities can be used to fabricate a nuclear weapon or other nuclear explosive device. The inspection activities are designed and implemented so as to detect abrupt diversions of one 'significant quantity' or more during each successive one month period, and protracted diversions of one 'significant quantity' or more during each successive one year period. One 'significant quantity' is defined in Section 9.3.

Safeguards experience in reprocessing, in conversion and MOX fabrication, and in plutonium fuelled reactors, has matured to the point that the IAEA is able to derive conclusions regarding such operations and the plutonium flows and inventories encountered. IAEA safeguards cannot determine whether or not a State might harbour nuclear weapon ambitions. Also the IAEA is not entitled to restrict the accumulation of separated stocks of any safeguarded materials (plutonium, highly enriched uranium (HEU), low enriched uranium (LEU), etc.) that could be converted to nuclear explosives in a relatively short period (which would depend on the effort the State is willing to exert, should the State so determine). The IAEA can, however, provide assurance at periodic intervals that the amounts of plutonium declared are consistent with expectations, and that all declared plutonium remains accounted for and committed to peaceful use.

Over the past 30 years the international non-proliferation regime has grown to include treaty obligations, controls on nuclear commerce and verification. The IAEA safeguards system has continued to evolve during that period to address increasingly complex facilities and, following the revelation of Iraq's clandestine nuclear weapons programme (which used HEU), to include concerns for undeclared nuclear materials and nuclear operations in violation of the NPT and IAEA safeguards undertakings.

As a result of the Iraqi revelations, comprehensive IAEA safeguards agreements were extended to provide legal rights for ensuring that all nuclear material and all relevant nuclear operations are declared and subject to inspection. The mechanism for this expanded purview is the Additional Protocol [199], and efforts are underway to gain universal adoption of this fundamental measure. The non-proliferation regime continues to evolve to keep pace with new developments in the industry and elsewhere, and further effort will no doubt be required if plutonium utilization is to expand significantly.

In this section, in addition to a review of security and physical protection issues, the current status of safeguards implementation at MOX fabrication facilities is reviewed as an example of the implications of effective safeguards (at least for that portion of the chain). The implementation of safeguards at LWRs that use MOX is not considered further here due to the fact that the techniques applied are, in principle, similar to safeguards at LWRs that do not use MOX [200, 201]. For MOX fuelled

LWRs, the verification activities are merely reinforced so as to maintain the continuity of accurate knowledge on fresh MOX during shipment to, and storage at, the reactor site. The permanent monitoring and enhanced surveillance are maintained up to the loading of each MOX fuel assembly in the core; final confirmation that the MOX assemblies have not be removed is made before the core is closed. Should continuity of knowledge be lost for one or more MOX fuel assemblies, safeguards inspectors conduct at the reactor site non-destructive assay (NDA) verification of the plutonium content and the isotopic composition in this (or these) fuel assembly (assemblies).

9.2. THE AUTOMATION OF MOX FUEL FABRICATION AND THE EVOLUTION OF IAEA SAFEGUARDS TECHNOLOGY

At present, there are three major MOX fuel fabrication facilities in operation producing LWR fuel (MELOX and Cogema/CFCa in France and Belgonucleaire in Belgium), a fourth in commissioning (SMP in the UK) and design activities are underway for a fifth (J-MOX in Japan). A major facility for fast reactor fuel is also in operation (PFPF in Japan). In the context of the excess weapons plutonium disposition programme, major MOX facilities are also being planned at Savannah River in the USA and Ozersk or Krasnoyarsk in the Russian Federation.

Each new facility constructed has introduced increased automation. In addition to normal industrial motivations of increasing productivity while reducing the number of plant operating staff, MOX plants have been driven by the desire to limit radiation exposures and to improve product quality control by means of 100% examination of the fuel products at key steps in the manufacturing operations. Distributed processors are used to control operations with a degree of reliability that humans are incapable of providing. Use of remote measurement equipment and other techniques provide a means for ‘hands-off’ operations with unparalleled product quality and minimal scrap rejection.

One of the requirements and benefits of automated processing is continuous, positive control over the flow of material. Process equipment is increasingly designed to prevent the accumulation of fissile material, especially powder, within and around the process equipment [42, 202]. This results in minimal process hold-ups, which might otherwise decrease safeguards capabilities and raise costs and radiation hazards in order to maintain the unmeasured working inventory (the so-called ‘material unaccounted for’) within acceptable limits.

During this same period, the tendency in safeguards has been towards increasing use of unattended measurement and monitoring systems [203–206]. Each system is designed using advanced computational techniques to optimize the sensitivity and uniformity of measurements. The measurement systems are integrated with containment and surveillance systems so as to provide assurance that all items passing a certain point in the process are measured once and only once, that the measurements are carried out under specified and controlled conditions and that the measurement data are captured, stored and evaluated in such a manner as to ensure their authenticity. The monitoring systems provide a capability for maintaining continuity of knowledge of the verified items.

Important to the design of safeguards concepts and the safeguards measures to be applied at modern MOX fuel fabrication plants is the fact that they are designed as ‘hands off’ facilities and, as such, are more or less fully automated. To remove material from the normal material flow only for the purpose of verification, therefore, is either not possible or, in the case of facility-specific radiation protection requirements, for example, very undesirable for the operator. As a consequence of the move towards automation in MOX plant operation, the development of safeguards systems verifying the material flow and inventories needs to move towards automated inline/online equipment operating in unattended mode. The goal here is to reach, via a combination of NDA equipment, monitoring systems and the application of containment/surveillance (C/S) measures, a maximum amount of nuclear material which can be regarded as being continuously verified while, at the same time, reducing the inspection effort [207, 208].

Increasingly, such systems are being installed at key locations in the process and at the entries and exits from stores within the facility. Furthermore, the measurement data and the C/S data now tend to be remotely transmitted to a central collection point within the facility where they can be archived and reviewed on-site. At the PFPF facility in Japan, for example, progress is being made on introducing remote monitoring concepts to transmit unattended measurement and monitoring data to an IAEA Regional Office [209, 210]. Through such means, the IAEA is able to apply more effective safeguards than it could through other means, with less inspection effort than would otherwise be required.

9.3. BASIS OF THE SAFEGUARDS CONCEPT: OBJECTIVES AND BOUNDARY CONDITIONS

The objectives and methods of IAEA safeguards are set forth in INFCIRC/153 [211] and in the IAEA Safeguards Agreements based upon that document. Following the detection of Iraq’s clandestine weapons programme, the IAEA safeguards system has been strengthened by adding features that were determined to be within the purview of these comprehensive IAEA Safeguards Agreements based upon INFCIRC/153. The safeguards system was further extended by introducing an Additional Protocol based upon INFCIRC/540 [199], which provides the IAEA with the legal right to verify the absence of undeclared nuclear material, activities and related process facilities. At present, a substantial effort is being made by the IAEA and by the Euratom Safeguards Office to integrate the INFCIRC/153 and 540 verification methods for optimum effectiveness. Provisions governing facilities that process, store or use separated plutonium will be established following the completion of arrangements on less sensitive materials.

These safeguards objectives and methods are further specified in the Safeguards Criteria. The key considerations published by the IAEA and contractually specified in the Safeguards Agreements are:

(a) Timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection;
(b) Material accountancy as a safeguards measure of fundamental importance, with containment and surveillance as important complementary measures.

The safeguards requirements for MOX fuel fabrication facilities within the overall safeguards system are determined by the inspection goals for “unirradiated direct-use material” and described in the Safeguards Criteria as follows:

(a) The *significant quantity* (SQ) is defined as the approximate quantity of nuclear material in respect of which the possibility of manufacturing a nuclear explosive device cannot be excluded. One SQ of plutonium is defined to be 8 kg.
(b) The general definition of the *timeliness* component of the inspection goal is that the safeguards activities for detecting anomalies that could indicate abrupt diversion of one SQ during the timeliness period applicable to the nuclear material have been carried out and conclusions drawn on the occurrence of such anomalies. Conclusions related to abrupt diversion scenarios are required at monthly intervals.

During the regular (monthly or continuous) inspections for maintaining timeliness, PuO_2, MOX powder, pellets and scrap are to be verified with medium detection probability for gross and partial defects. Fuel rods, fuel assemblies and other fuel items, as well as other bulk material (e.g. waste) containing plutonium, are to be verified with medium detection probability (i.e. 50%) for gross defects.

Once a year, in the annual physical inventory taking, PuO_2 and MOX powder and other homogeneous bulk materials are to be verified with a high detection probability for gross, partial and bias defects and to detect a protracted diversion of one SQ over a longer period of time. For fuel rods, fuel assemblies and other fuel items, verification with a high detection probability for gross and partial defects is required. Other bulk material such as waste needs to be verified for gross defects. These requirements are also valid for the verification of receipts and shipments.

9.4. NUCLEAR MATERIALS ACCOUNTANCY

The essential basis for national controls on nuclear materials, and the corresponding basis for the IAEA (and Euratom) safeguards implementation in a MOX facility, is nuclear material accountancy. Modern information systems allow significant enhancements of the operator's materials accountancy system and in some cases have been used to develop and implement near-real-time material accountancy [212]. From experience, the more detailed and accurate the accounting data are, the less effort needs to be spent by the operator on the preparation of the monthly interim inventory verifications (IIVs) and on the physical inventory verification (PIV), and the less intrusive safeguards will be for normal production.

It is the operator's material control and accountancy system that provides the data declared by the operator and the subsequent State declarations, which are to be verified independently by the IAEA. The verification activities include examinations of operating and accounting records and, on a statistical basis, independent measurements (NDA and destructive analysis (DA)) to check the correctness of data given by the operator. C/S measures substantially reduce the number of measurements necessary for maintaining the continuity of knowledge of verified materials.

9.5. SAFEGUARDS TOOLS

From the most traditional accountancy verification by DA to the most elaborate remote monitoring technique, many of the advanced safeguards tools were developed for use in MOX fabrication plants. There is continuing evolution towards increasing effectiveness and efficiency in safeguards implementation.

9.5.1. Destructive analysis

The most accurate measurements attainable serve as the foundation for nuclear materials accountancy. Destructive analyses are required on shipments and receipts of plutonium in bulk form, and on inventory materials at MOX plants mainly applied during the annual PIV. DA samples are taken from materials in the bulk handling stage, usually powders, pellets and conditioned clean residues.

Such measurements include bulk determinations by weighing plutonium bearing materials, or through volume measurements, together with sampling for laboratory analysis. Sampling is carried out following procedures that ensure that the samples taken are representative. In some cases, in order to limit the amounts of plutonium shipped to the IAEA Safeguards Analytical Laboratory (SAL), the plutonium samples are dissolved, aliquoted, spiked (immediately or later at the SAL) and dried for shipment. Such samples generally contain less than 10 mg Pu per sample, an amount adequate for very high accuracy measurements at SAL.

To minimize the transport of nuclear material, especially in the modern industrial scale plutonium handling facilities (reprocessing and MOX fabrication) in Europe, Euratom has adopted on-site laboratories. For the Rokkasho-mura site in Japan, the IAEA also plans an on-site laboratory.

Over the years, a variety of wet chemical analytical methods have been applied to determine the plutonium content of samples. In recent years, however, the use of isotope dilution mass spectrometry (IDMS) methods has been widely adopted. IDMS is easier to use in a production environment and produces consistent state-of-the-art results [213].

9.5.2. Non-destructive assay

NDA measurements are made on all material flows and inventories of plutonium encountered in MOX fuel fabrication facilities. Two primary methods have evolved [214–217]:

(1) High resolution gamma ray spectrometry (HRGS) is used to determine the relative concentration of the plutonium isotopes and the concentration of ^{241}Am in all forms of plutonium. Such measurements are used either to confirm more accurate mass/alpha spectrometric measurements made on laboratory samples or to establish the isotopic composition — especially for heterogeneous materials. High purity germanium detectors are widely used. These detectors require cooling with liquid nitrogen to provide the energy resolution required. Special electrically powered refrigerators are now being introduced, which will make it more practical to employ HRGS systems in unattended, continuous operation monitoring systems.

(2) Neutron coincidence measurement systems have been adapted to virtually every plutonium configuration encountered in MOX plants, including small samples containing 1 g or less of plutonium, any size of feed and intermediate products, FRs, FAs waste containers and distributed hold-up in process equipment. The technique is based upon the fact that, in a fission event, two, three or more neutrons are emitted within a brief period. Coincidence detection systems measure additional neutrons following an initial neutron detection event, within the time required for the additional neutrons to travel to the detectors (normally ^{3}He detectors). When the plutonium is well controlled, as in most cases within a MOX plant, the detection of coincidence neutrons provides a means to measure the ‘effective ^{240}Pu mass’ to within 3% (or better) accuracy. The effective ^{240}Pu mass is calculated from the isotopic composition, reflecting the spontaneous fission yields from ^{238}Pu and ^{242}Pu, in addition to

the ^{240}Pu. For unattended use, the signals emerging from the detectors are split and processed through redundant shift registers and dedicated computers. Using this method, unattended plutonium assay systems have operated for 11 years in one facility without a single system failure, and, according to the operator, have saved substantial amounts of money and radiation exposure compared with the use of attended verification activities. In those cases where the properties of the material are not closely controlled, measuring the triple coincidence rates offers a practical means of obtaining high accuracy measurements. Heterogeneous scrap can be measured more accurately through this means than by any other practical method. The limiting error sources affecting overall plutonium determinations by HRGS and coincidence counting are being examined to determine whether accuracies better than 1% can be attained. In MOX facilities, variable moisture content is the principal source of error. An effort is underway to redesign measurement systems for feed powders to meet that capability, which will then result in savings in time and money for additional sampling and laboratory analyses.

Unattended assay and monitoring systems are preferable as they take place during routine operations in MOX plants in a non-intrusive fashion. Moving an item from a production line to a separate measurement station is often not possible and space/measurement time limitations may require integration of measurement devices into installed operator measurement/handling posts.

Gamma and neutron systems are employed in MOX plants at numerous positions, including the monitoring of receipts and movements of plutonium cans, pellet trays, FRs and FAs and wastes. They can be mounted on materials handling machines directly or positioned in the normal process flow route so as to avoid additional operations. With the introduction for routine use of the MGA evaluation code for gamma spectra [218], which allowed the reduction of measuring times to a few minutes, and the use of powerful processors for merging and evaluating measurement data, a new generation of integrated measurement stations has been created.

9.5.3. Containment and surveillance

C/S measures are well established complementary methods. Typically these involve techniques such as video camera surveillance and a range of barrier systems often incorporating tamper indicating seals. Camera systems may be integrated into, and triggered by, other equipment such as movement detectors, radiation detectors or electronic seals to improve the efficiency of the review process.

Multiple C/S systems are used at significant intermediate stores in MOX plants to avoid loss of continuity of knowledge and the corresponding need for additional verifications. C/S areas also include PuO_2 feed stores and finished fuel stores.

9.5.4. In-process monitoring

Branching of an operator's own flow measurement and monitoring equipment is implemented by using a mix of independent sensors, electronics and software, which enables inspectors to authenticate the operator's declaration by analysis of raw signals from the plant.

In the new MOX plants, branching is most widely used on weighing equipment and automated identification systems for PuO_2 cans and MOX powder containers, pellet trays, FRs and FAs. In-process monitoring, together with increased transparency, leads to more effective safeguards on the nuclear material in the facility with lower operator and inspector costs, and far less inspector presence, than would otherwise be required.

Process hold-up has been, and remains, a significant challenge in MOX plants. The process equipment and surrounding surfaces must be, if they are not already, engineered so as to minimize opportunities for plutonium powders to spread and accumulate. Apart from that, NDA methods are now also available for in situ verification of hold-up.

9.6. SECURITY AND PHYSICAL PROTECTION

National systems must protect against a broad range of threats, including unauthorized use and theft by insiders or outside forces. National systems include material protection, control and accounting to address all such concerns.

Whereas 'Safeguards' concentrates on regulating the control and accounting of nuclear material, 'Physical Protection' focuses on protection of nuclear material from theft (or unauthorized diversion) and sabotage for both non-proliferation and radiation safety purposes. Physical protection is clearly a matter of international concern and co-operation. Thus, while responsibility for establishing and operating a comprehensive physical protection system for nuclear materials and facilities within a State rests entirely with the government of that State, transport across international borders, for example, has required the establishment of internationally agreed standards and obligations.

A hierarchy of documentation exists to provide a system of legal requirements as well as recommendations for good practice and other general guidelines. These are summarized in Sections 9.6.1–9.6.3.

9.6.1. Legal requirements

The only legally enforceable obligations are those on international movements of nuclear material. The Convention on the Physical Protection of

Nuclear Material (CPPNM) [142], which entered into force in 1987, obliges signatories to:

- Make specific arrangements and meet defined standards of physical protection for international shipments of nuclear material,
- Co-operate in the recovery and protection of stolen nuclear material,
- Criminalize specified acts or threats to misuse nuclear materials to harm the public,
- Prosecute or extradite those accused of committing such acts.

The Convention also promotes international co-operation in the exchange of physical protection information. Domestic use, storage or transport are not covered.

9.6.2. IAEA standards and recommendations

Underneath the Convention's legal requirement there lie a series of standards and recommendations developed by the IAEA which cover physical protection for nuclear material in use, storage and transport, both domestically and internationally. These standards are embodied in INFCIRC/225/Rev.4 (The Physical Protection of Nuclear Material and Nuclear Facilities) [141], which, although receiving widespread international recognition and implementation, has no legal standing.

These standards differentiate between what individual states should be addressing and what the IAEA aims to provide. The State's physical protection system should establish conditions which minimize the possibilities for unauthorized removal of nuclear material and/or sabotage, as well as provide measures to locate and recover missing material and minimize any radiological consequences of sabotage. The objectives of the IAEA are limited to providing recommendations and advice for consideration by States, recognizing that States are responsible for their own systems of physical protection.

The standards describe elements that a State's national system of physical protection should address and highlight recommended measures in each area. This includes aspects such as appropriate legislation and regulations, confidentiality and regular evaluation as well as more specific concepts such as hardware (security devices), procedures (including the organization of guards) and facility design (including layout).

In determining the level of physical protection to be implemented for nuclear materials in use, storage or during transport, account needs to be taken of the possibility that material could be diverted for the purposes of constructing a nuclear explosive device. To that end, nuclear material is categorized on the basis of the potential risk of it being used for such a purpose. Considered in this categorization are such aspects of the material as type (e.g. uranium or plutonium), isotopic composition (content of fissile isotopes, but not for plutonium), physical and chemical forms, degree of dilution, radiation level and quantity. For instance, material classified as Category I (e.g. over 2 kg of plutonium) should be controlled within a defined high security ‘inner area’; Category II material (e.g. between 0.5 and 2 kg of plutonium) need only be controlled within a ‘protected area’ for which slightly different security requirements are defined.

With regard to sabotage, appropriate measures for protection are based on potential off-site radiological consequences. Nuclear material, systems or equipment, the sabotage of which could lead to unacceptable radiological consequences, should be located in defined ‘vital areas’ for which appropriate levels of guards, response forces and access are specified.

9.6.3. Guidelines for implementation of standards

Supporting INFCIRC/225/Rev.4 is IAEA-TECDOC-967 (Rev.1) [219], which provides guidance and considerations for a State’s competent authority to understand and prescribe appropriate requirements consistent with INFCIRC/225/Rev.4. Reference [219] lists each clause of INFCIRC/225/Rev.4 and provides explanations that indicate how a State may go about interpreting the recommended standards. It outlines, for instance, what approaches may be used to reach a conclusion about the categorization of nuclear material, or what issues should be borne in mind when defining access points to ‘inner areas’ or ‘protected areas’.

In addition, it is worth noting that a Physical Protection Handbook is in preparation by the IAEA which will provide further guidance to the licensee or designer of physical protection systems who may have specific implementation and compliance responsibilities.

9.7. FUTURE DEVELOPMENTS

Computer hardware and software are used to a large extent for performing safeguards inspection activities at all stages. The implementation of advanced systems for nuclear material accountancy, improved material verification equipment and enhanced data evaluation methods require a new, standardized and uniform approach regarding safeguards data processing. The IAEA is currently developing a complex computer software model for a MOX fuel fabrication facility that will integrate data collection, evaluation and reporting for safeguards inspection purposes in a modular structure using a database system [220]. This will also meet the requirements of the Additional Safeguards Protocol.

In recent times, a variety of advanced unattended systems for remote monitoring and transmission through telephone lines or other secured data transmission channels have been developed. The instruments used in such applications include digital cameras, electronic seals, and radiation monitors and sensors. Remote monitoring offers the possibility of replacing many of the on-site activities that involve unattended assay and monitoring systems with data collection, review and evaluation at a remote location, in essentially real time.

Despite the progress already made towards automation of safeguards, introducing plutonium fuels into a State will require a much higher level of safeguards involvement with the State. For example, an LWR without MOX requires 8–10 days of IAEA inspection per year, a number that will be reduced when integrated safeguards are implemented, reflecting the introduction of an Additional Protocol within the State. With MOX fuel, the inspection activities may increase up to 50 inspection days per year, not taking into account possible reductions under integrated safeguards. MOX fabrication plants currently require up to 200–300 inspection days per year, and reprocessing plants of the order of 1000 inspection days per year. However, a significant reduction in the work of field inspectors may be expected following moves to automated safeguards equipment in an unattended mode, remote monitoring and remote data transmission.

Designing safeguards into a new facility at the outset will ensure that the IAEA will be able to apply effective safeguards at the lowest cost to the international community, and also provide the highest level of assurance of non-proliferation linked to these facilities. In this regard, the continued co-operation among States, facility operators and the IAEA remains essential.

9.8. CONCLUSIONS

For twenty years, the utilization of plutonium has been impeded by high costs and the concerns of the international community regarding the suitability of plutonium for use in nuclear weapons or other nuclear explosive devices. During that period, the IAEA, Euratom and the industry have continued to develop improved plants and improved safeguards. The improvements were most often complementary — automating plants led to improved product quality and uniformity, which permitted improved diversion detection sensitivity. Integrating safeguards equipment into the material transport systems within a plant and sharing the use of key operator equipment have allowed the IAEA and Euratom to provide better coverage at lower cost and with significant reductions in intrusion into normal plant operations.

This success has been evolutionary. In this information age, great advances have been made in automating plant operations to improve worker productivity and to minimize risks where hazardous materials or operations are involved. So too, in safeguards, where acquiring pertinent data in real time and transmitting it to a central location within a facility has made it possible to track operations more closely and to use modern information analysis methods as a means to improve the timeliness and sensitivity of IAEA safeguards further. The safeguards system designed for each new MOX plant builds on the experience gained at all previous plants.

During the last decade there has been a steady increase in the number of unattended verification and monitoring systems, which has resulted in an increase in the verification coverage and, at the same time, reduced on-site inspections. These automated systems operate in the unattended mode and combine surveillance, NDA techniques and/or process monitoring devices. The advantages of such systems are reduced inspection effort, 100% verification with reduced levels of intrusiveness to the operation of the nuclear facility and reduced radiation exposure to inspectors and plant operating staff. Such assay and monitoring techniques are especially well suited to automated and remotely operated facilities.

10. SPECIFIC ASPECTS OF FAST REACTOR MOX FUEL

10.1. INTRODUCTION

It is generally agreed that the major mission of fast reactors (FBRs), breeders and burners, is to provide a long term stable energy supply and that the breeding of plutonium has an essential role to play in this. However, circumstances and views on some key questions, such as the time needed to introduce FBRs commercially, the doubling time needed to meet the limitation on the natural uranium consumption, and the FBR economics, vary from country to country. As a result, some countries have stopped or postponed fast reactor development programmes (political reasons have had a significant influence on these decisions), whereas activities on FBR systems are still ongoing in China, France, India, Japan and the Russian Federation.

Originally, the principal goals of FBR design were short doubling time and a major cost reduction. The requirement on doubling time affects FBR design because most of the accompanying design objectives to achieve low doubling time and low cost were conflicting. Looking at the current energy situation and future perspectives, an incentive to shorten the doubling time at the expense of cost is unlikely to be found. Therefore, it is clear that the previous principal goals of FBRs need to be modified to new goals that fit with current visions. The new goal of FBRs could be to provide an almost inexhaustible and economically competitive energy source, which also satisfies requirements from environmental aspects and non-proliferation of nuclear material. For example, the breeding ratio should be such that only quantities of fuel which are needed for prevailing consumption requirements are generated, minimizing the flow of material within the nuclear fuel cycle and giving consequential benefits for economics, ecology and non-proliferation. A revision to the design concept of the FBR is needed in order to accomplish this new goal.

Besides the major mission mentioned above, an FBR system can be applied flexibly to diverse fuel types and compositions, including plutonium and minor actinides (although not necessarily all isotopes at present). This suggests that the technology could contribute to a solution to waste problems by reducing the amounts of transuranic materials that remain in high level radioactive waste and constitute high potential risks. In addition, a combination of an FBR system with the existing LWR systems may provide a more favourable and flexible system from the economic and environmental points of view. Some new development programmes have commenced from these viewpoints.

This section concentrates on aspects of MOX fuel in FBRs both past and present [6]. At the end of this section, future possibilities of FBR development are presented.

10.2. CHARACTERISTICS OF FBRs

To provide the flexibility for operation as a breeder or a burner, the core of an FBR has to have the capability to operate at high heat ratings and achieve high burnup in order to minimize fuel doubling times, and be as compact as possible to minimize fuel inventory costs. Thus, in the case of fast reactors, the fuel element design has a major impact on the overall performance capabilities of the reactor. The specifications of a compact core with high heat ratings require the placing of the fuel in small diameter pins separated from one another by narrow coolant channels. This configuration led to the choice of liquid sodium as the coolant for virtually all FBR designs.

With these requirements in mind the fuel element specification has to account for the following:

- high linear power / heat rating / surface heat flux
- high burnup
- low distortion from swelling, bowing, etc.
- no pin-to-pin failure propagation.

The fuel material has to have:

- high fissile atom density
- good thermal conductivity
- good compatibility with fuel cladding and reactor coolant
- as few moderating atoms as possible
- low swelling from fission products

- high melting point
- good resistance to radiation damage.

The cladding has to have:

- good creep and yield strength with retention of adequate ductility
- low swelling at high fast neutron doses
- good compatibility with the fuel and reactor coolant.

The initial FBR development programmes were based on the use of metallic fuels, which were later followed by an emphasis on ceramic fuels and, in particular, on MOX. During the development of ceramic fuels for FBRs, a number of fuel forms were tried in order to accommodate fuel swelling — for example solid pellets with normal porosity, hollow (annular) pellets, dished ends and vibro-packed powder. Solid or annular pellets helium bonded to a stainless steel cladding emerged as the preferred fuel element form.

The remainder of this section concentrates on oxide FBR fuels only. For an assessment of alternative fuel types, such as nitride fuels and inert matrix fuels, Section 11 should be consulted.

10.3. HISTORICAL BACKGROUND

This section discusses the use of oxide fuel in experimental reactors, prototype fast reactors and commercial fast reactors [221–229].

10.3.1. Experimental reactors

The first recorded use of oxide fuel in a fast reactor was in the Russian 5 MW BR5 reactor, which went critical in 1959 with a core of PuO_2 that was taken to 5% fima (fissions per initial metal atoms) burnup (≈45 MW·d/(kg HM)), but with numerous fuel failures. In 1965, the core was replaced with uranium carbide pellet fuel. The power was then raised to 10 MW (BR10) and three more cores were consecutively tested in the reactor, two cores with PuO_2 fuel and one with uranium mononitride. A maximum burnup of about 14% fima (≈130 MW·d/(kg HM)) was reached in updated designs of plutonium dioxide fuel rods.

In 1968 the experimental reactor BOR-60 started operation at RIAR, Dimitrovgrad, and systematic studies of MOX fuel followed. Since then, sixteen pelletized fuel assemblies and 450 MOX vibro-packed fuel assemblies have been irradiated in BOR-60.

France took the decision to use MOX fuel as the driver fuel for its fast reactors in 1963 and Rapsodie, a 20 MW reactor designed to test MOX fuel assemblies, began operation in 1967. The experimental programme confirmed the feasibility of reaching high burnups and high damage doses with MOX fuel in stainless steel cladding.

In the UK, a number of test irradiations of MOX fuel were carried out in the 15 MW DFR, which operated between 1959 and 1977, with a driver charge of U–Mo metal fuel. The success of these irradiations, and the experience gained at that time with thermal reactor irradiations of ceramic fuel, led to the decision that the PFR to be built at Dounreay should have a driver charge of MOX fuel.

In Japan, the experimental fast reactor Joyo is operating with MOX fuel. It first reached criticality with the Mk-1 core in 1977. In 1983 the reactor power was uprated from 75 to 100 MW with the Mk-2 core. A number of experiments have been conducted in the reactor including, in 1984, the irradiation of fresh MOX fuel fabricated using plutonium recovered from reprocessed Joyo spent fuel, an experiment which demonstrated the successful closing of the fuel cycle. The maximum burnup achieved with the Mk-2 core was 75 GW·d/(t HM).

MOX fuel was also utilized for the German 58 MW KNK-II fast reactor, which operated at Karlsruhe from 1977 to 1991. The maximum burnup achieved in this reactor was around 100 GW·d/(t HM).

10.3.2. Prototype fast reactors

Four prototype fast reactors (BN-350, Phénix, PFR and Monju) have been constructed and operated throughout the world. A fifth, the German reactor SNR-300, was also constructed but never operated. Its contribution to MOX fuel development was in the fabrication area.

BN-350 operated in Kazakhstan from 1972 until it was closed at the end of the 1990s. Uranium dioxide fuel was used as the driver charge fuel for the reactor, but fuel assemblies with MOX fuel have also been tested. In 1982 ten fuel assemblies with pelletized MOX fuel were loaded and in 1988 a further four assemblies were manufactured and loaded. From 1985 until 1987 two assemblies containing vibropacked MOX fuel were also irradiated. Following the plant closure, all the fuel has now been discharged.

Phénix has operated at Marcoule since 1973 and it is planned to continue operation until 2004. The reactor characteristics are very similar to those of PFR. Between 1973 and 1990, Phénix provided valuable experience on standard and experimental MOX fuel. Phénix fuel has been reprocessed along with thermal reactor fuel. The recovered plutonium was used to fabricate fresh MOX, which was then irradiated in Phénix, thus completing the fuel cycle.

PFR (250 MW) operated at Dounreay from 1974 to 1994 with MOX annular pelletized fuel as the driver charge. A large number of experimental fuel designs were tested in PFR which explored variations in parameters such as ratings, pellet diameters and cladding materials. In late 1982, fresh MOX fuel fabricated using plutonium recovered from spent driver charge fuel was loaded into the PFR core, thus also completing the fuel cycle.

Monju (714 MW) [230] achieved first criticality in 1994 and operated at 40% of its rated power until 1995, when a sodium leak occurred. Although the schedule is not finalized, the preparation for modification of Monju for safety enhancements is under way.

10.3.3. Commercial fast reactors

The Russian commercial fast reactor BN-600 started operation in 1980 and is still operating today. The driver fuel is uranium dioxide, as in the case of BN-350, but the core also contains a number of MOX fuel assemblies. Twenty assemblies containing pins with pelletized MOX fuel and six assemblies with vibro-packed MOX fuel have been successfully irradiated to date. Sixteen additional assemblies containing MOX pellets were fabricated in 1999 and 2000 and have been loaded.

Superphénix operated at Creys-Malville from 1985 to 1996 but ceased operation as a result of a decision by the French government, after 320 effective full power days.

10.4. PIN AND ASSEMBLY DESIGN

The fuel pin for a MOX fuelled FBR consists of an axial stack of cylindrical core fuel pellets, with axial blanket pellets at each end of the fuel column, the whole encased in a cladding; fuel pins are bundled to form a fuel assembly. This basic structure is similar to that of LWR fuel rods.

However, FBR MOX fuel has a number of characteristics that differ from those of LWR fuels, which result from the design requirements unique to the FBRs, including the hard neutron energy spectrum, compact core size, high power density, high burnup, high temperature and plutonium breeding. The major differences from LWR fuel designs are listed below:

(a) A high plutonium content, in order to have sufficient reactivity in a compact core with a high energy neutron spectrum;
(b) A low smeared fuel density, to suppress increases of pin outer diameter due to fuel swelling at high burnup;
(c) The use of an axial blanket of depleted uranium dioxide to breed plutonium;
(d) A long gas plenum, comparable to the core pellet column length, in order to accommodate the fission gases produced and released to the free volume of the fuel pin during irradiation to high burnup and temperature;

(e) A small outer diameter of fuel pin to accommodate a high power density;

(f) A hexagonal fuel pin arrangement with small gaps provided by a wire spacer helically wound around each of the fuel pins or by a hexagonal grid spacer;

(g) A wrapper tube surrounding each fuel bundle in order to form a sodium flow channel for efficient cooling and to prevent fuel failure propagation under accident conditions;

(h) The use of austenitic or ferritic steels or nickel alloys as materials for structural components in order to have good compatibility with sodium and to cope with high temperatures and high levels of fast neutron exposure;

(i) A high coolant heating rate, namely more than 200ºC/min compared with around 10ºC/min in LWRs.

The fuel structure and actual fuel design varies with the reactor scale, design targets and the design methodology. Large scale reactors, for instance, have lower plutonium fissile content and greater outer diameter of fuel pin than small scale reactors. For burnup extension, the gas plenum needs to be extended, the smeared pellet density lowered and a long life material used for structural components. Alternative fuel pin options to fit with design requirements exist, such as:

(1) A low density solid pellet or a high density annular pellet to lower the smeared fuel density.

(2) An upper plenum configuration, with the advantage of no fission gas passing through the core in the case of cladding rupture, or a lower plenum configuration, with an advantage in minimizing plenum length as shown in Fig. 27.

(3) A homogeneous core pin configuration, with one stack of core pellets surrounded by upper and lower axial blankets, or a heterogeneous core pin configuration, with an internal axial blanket region, separating the core pellet region into two parts as shown in Fig. 27.

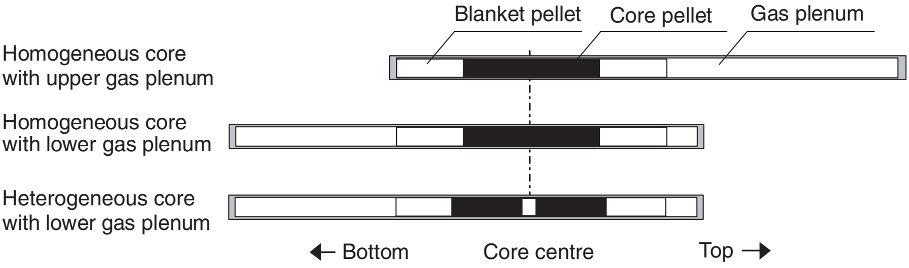

FIG. 27. Axial configuration of an FBR fuel pin.

An example of an FBR fuel assembly structure (Monju) is given in Fig. 28.

10.5. FUEL PERFORMANCE ISSUES

The severe operating conditions in FBRs mean that special considerations for the fuel behaviour and consequences are necessary. This section discusses MOX fuel behaviour in FBRs, the irradiation experience attained and the capabilities of MOX fuel inferred from this experience [230–235].

10.5.1. Fuel restructuring during irradiation

Conditions in fast reactor cores cause restructuring of the MOX fuel to form four distinct regions at linear heat generation rates of 40 kW/m and above.

The innermost region is a central void that results from transport of as-fabricated porosity and some of the fuel cladding gap up the temperature gradient to the fuel centre.

The fuel surrounding this void consists of dense grains, of at least 98% theoretical density (TD), that are elongated radially. These grains form lenticular voids that move inwards by fuel vaporizing from the hotter (inner) side of the voids and condensing on the cooler (outer) side, giving a net outward movement of fuel. The lenticular voids develop from large (>5 µm) fabrication pores or startup cracks in the fuel, which heal in the process. Zones of increased plutonium content may be observed within this columnar grain region as well.

Outside the columnar grains is a region where temperatures are sufficiently high for grain growth to take place by bulk diffusion and for some biased movement of voids and inclusions to occur up the temperature gradient by surface diffusion. The enlarged fuel grains in this region, although normally termed ‘equiaxed’, are slightly elongated in the direction of the temperature gradient with their boundaries invariably decorated with gas bubbles and fission product inclusions.

The fuel between the equiaxed grain growth region and the cladding retains its original microstructure and density and is simply labelled the unrestructured region. Fuel in this region operates at temperatures below about 1200°C, where mobilities are low and where, therefore, the fuel tends to retain most of its original characteristics. However, radiation does enhance bulk diffusion and creep rates in this region, so that limited hot pressing and oxygen redistribution may take place.

10.5.2. Pin irradiation experience

Over 265 000 pins containing MOX have been irradiated in European fast reactors. In the Russian BN-350 and BN-600 reactors 1500 and 2500 rods,

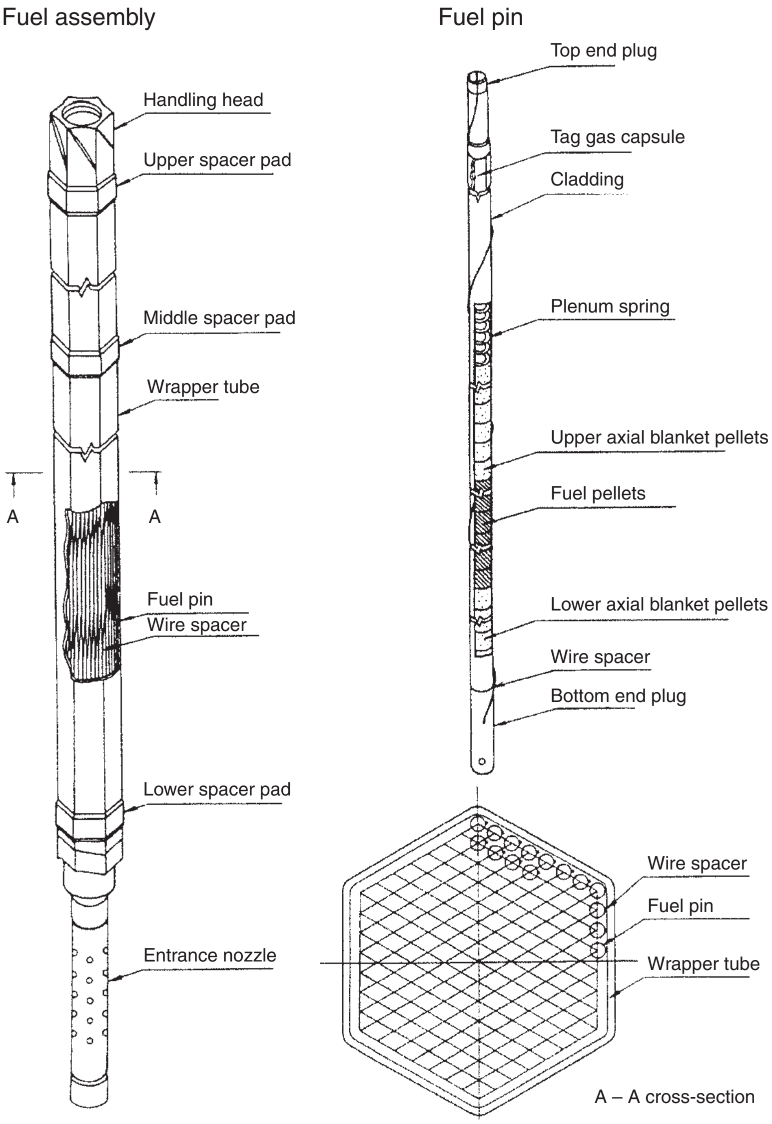

FIG. 28. Fuel assembly structure of Monju.

respectively, with pelletized MOX fuel have been irradiated. In addition, approximately 1000 rods with vibro-pack MOX fuel were irradiated in BN-350 and BN-600 at burnups up to 130 MW·d/(kg HM), and more than 16 000 rods with vibro-pack MOX fuel were irradiated in the BOR-60 reactor at even higher burnups. A further 54 000 have been irradiated in Japan. Of these, more than 61 Japanese (Joyo) pins have reached burnup values of 130 MW·d/(kg HM), though none have exceeded 150 MW·d/(kg HM). In addition, some experimental pins irradiated in test and prototype reactors throughout the world have attained burnup levels greater than 210 MW·d/(kg HM).

10.5.3. Fuel failure experience

It is important that, for safe reactor operation, the integrity of the cladding is maintained during irradiation. There are four main threats to the cladding integrity in FBR pins, namely manufacturing defects, mechanical interaction between pin, spacer and wrapper, fuel–cladding mechanical interaction (FCMI) and fuel–cladding chemical interaction (FCCI). While the first two have been identified as the cause in over 40% of the failures noted in European fast reactors [236], FCMI and, in particular, FCCI, have nevertheless been the subject of much study and discussion.

FCCI in FBR pins is caused by the migration of oxygen and volatile fission products to the fuel surface due to the severe operating conditions and generally low density of the oxide fuel. FCCI consists mainly of the oxidation of constituents of the cladding material, particularly chromium, in the presence of caesium, molybdenum, tellurium and iodine.

FCCI has been observed in a number of forms, with broad front oxidative attack being the most commonly observed mode. The extent of this type of attack is influenced by fuel and cladding temperature and oxygen potential but is not related to stress. The mechanism is generally believed to be a fission product assisted oxidation process with caesium the main catalyst, although the presence of tellurium has been identified in some cases. The extent of oxidative FCCI increases progressively with burnup but is not considered a life limiting feature since pin designs have taken the anticipated clad wastage into account.

It should be noted that the experience gained with fuel in Phénix, PFR, Joyo and FFTF since the 1970s, in which there have been only two failures attributed specifically to FCCI, has demonstrated that early fears of high failure rates due to FCCI have not been substantiated. This lower than expected incidence of failure has been attributed to the ability of the volatile fission products to buffer the oxidation of the cladding by absorbing the excess oxygen and forming oxides in the fuel–cladding gap. In most cases the potential life limiting effects of FCCI have been accommodated within the fuel design and/or the reactor operating envelope.

FCMI has never been the major issue in FBRs that it has been in LWRs. This is mainly due to the fact that the fuel pin cladding is subjected to irradiation damage (Section 10.6). The consequent void swelling and irradiation creep ensures that the cladding moves away from the fuel during irradiation, thus avoiding any gross mechanical interaction.

10.5.4. Irradiation capabilities of MOX fuel

The experience gained from irradiation experiments and post-irradiation examination has led to the conclusion that the behaviour of MOX fuel is not a life limiting factor for fuel pins, even under the stringent high burnup requirements. The main observations confirming this point are:

(a) A moderate fuel swelling rate, even at very high burnup, and no dramatic degradation of heat transfer properties.

(b) Problems of mechanical and chemical interaction between the fuel and cladding can be overcome with adequate pin design. Recommended variations include using a moderate linear rating and increased pin diameter, resulting in a lower fuel surface temperature, a greater cladding thickness and a lower initial oxygen-to-metal (O/M) ratio (1.95–1.98).

(c) In the case of pin failure, no major reaction of fuel with coolant, which might lead to a great loss of fissile material or to a rapid development of secondary failures, or to failure propagation within the assemblies, have been observed.

On the basis of the excellent results obtained with MOX, it is currently believed that the target burnups for future large FBRs of 200 GW·d/(t HM) can be reached with this fuel type (subject to the limitations of the structural materials and the reactor control system).

10.6. FURTHER TECHNICAL ISSUES ASSOCIATED WITH FUEL IN FBRs

Besides the effects on fuel performance, the conditions experienced by fuel elements in fast reactor cores give rise to substantial damage to the crystalline lattice of the metallic materials used as core structural components. Considerable effort has been devoted over the last 30 years to understanding and improving the performance of these materials, since adverse behaviour can limit the efficiency and economics of the FBR system. Note, however, that the issues mentioned here are, in general, common to all types of FBR fuel and are not specific to plutonium bearing FBR fuels. Also, unlike thermal reactors, some of the problems highlighted can be mitigated in FBRs due to the higher temperature operation and the presence of a plenum in the fuel rods able to accommodate 100% fission gas release.

10.6.1. Types of radiation damage

The two forms of radiation damage are:

Atom displacement: When a fast neutron collides with an atom in a metal or alloy, the atom can be displaced from its equilibrium position in the crystal lattice. In turn, it can strike and displace other atoms. When the displaced atom comes to rest within, but not on, the crystal lattice, it is called an interstitial atom and the empty lattice position left is known as a vacancy. In fast reactor exposures, it is estimated that atoms in core materials will, on average, be displaced from their lattice sites more than 30 times a year. Reliable fuel operation up to 90 displacements per atom (dpa) has been justified in the Russian BN-600 design, corresponding to up to 400–450 effective full power days (EFPDs) operation.

Transmutation effects: The most serious of these are n–alpha, and possibly n–p, reactions, leading to the generation of helium and hydrogen in the metal matrix. At the very high neutron energies prevalent in fast reactors, these types of reaction are possible with virtually all metal atoms.

10.6.2. Effects of radiation damage

The effects of these damage mechanisms affect the mechanical behaviour of the material as they can give rise to embrittlement, irradiation creep and void swelling. Helium production causes metal embrittlement as the inert gas agglomerates into bubbles which migrate to grain boundaries; their growth under tensile stress can contribute to high temperature embrittlement.

Creep is a process that causes some materials under constant load to deform progressively over a period of time and can eventually lead to rupture. For a given load the creep rate increases as the temperature rises. Neutron irradiation enhances this deformation and, at the same time, induces creep deformation at temperatures where normal out-of-pile creep is negligible. This additional deformation mode makes extra demands on the ingenuity of designers in limiting loads to give acceptable deformations. On the other hand, there are situations where irradiation creep offers a valuable stress relief mechanism to reduce loads which might otherwise have built up to the fracture point.

In 1966 it was observed from irradiations in the DFR that considerable swelling of the cladding took place after prolonged exposure to neutron irradiation. Close examination of the irradiated cladding specimens revealed that very small voids or cavities had been formed within the grains. These voids, ranging from the smallest observable with an electron microscope up to 0.1 mm, did not contain sufficient gas to be called bubbles. They form only in the temperature range 350–700°C, which is precisely the operating temperature domain for FBR application.

The mechanism of void formation is relatively well understood. Dislocations in the regularity of atom planes act as sinks for both the interstitials and vacancies formed by atom displacement. However, the interstitial atoms have a greater tendency to migrate to the dislocations rather than the vacancies, i.e. the dislocations are 'biased' sinks. This bias gives rise to an excess of vacancies which, when stabilized by inert gas atoms such as helium produced by n–alpha reactions, can form three dimensional clusters. These, by the acquisition of more vacancies, can grow to form voids. The total volume change in the material depends on the fast neutron fluence or dose received and it is usual to relate this dose to the number of displacements per atom produced in the material.

The macroscopic effect of void swelling is that the metal swells and, as a result, fuel pins and the hexagonal wrappers, which constitute the assemblies, can increase in size both axially and radially. Volume increases of over 5% can be expected, which means that the linear dimensions can increase by about 1.5–2%. Unless sufficient clearance is allowed, coolant flow between pins may be reduced and, in the case of assemblies, removal or insertion during refuelling may be impeded. A more serious structural effect of void swelling arises from the variations in temperature and neutron flux that exist across the core of a fast reactor. Void swelling is sensitive to both temperature and damage dose and the variations in these across an assembly can cause the opposing faces to swell by different amounts, which results in bowing of the assembly. These problems can be solved by judicious choice of materials. Much effort has therefore been directed both in the UK and in other countries to the understanding of the phenomena involved, to the development of methods for predicting the changes induced and to the selection, optimization and proving of candidate alloys for FBR applications.

10.6.3. Fuel behaviour in transient and accident conditions

A number of experimental investigations into the behaviour of FBR fuel in accident or off-normal conditions have been carried out. The safe and reliable performance of MOX fuel was demonstrated in an extensive and broad based testing programme in the Experimental Fast Breeder Reactor II (EBR-II) in the Argonne National Laboratory (West) facility in Idaho [237]. A total of 57 tests were completed including 100% overpower transients, long term operation with multiple fuel failures and power-to-melt tests. These tests showed that MOX fuel pins can survive such off-normal conditions with little or no effect on their performance — in particular, breached blanket pins had no definitive effect on plant operations and the most limiting parameter for reliability was shown to be the integrity of the cladding properties.

Transient fuel pin behaviour was also explored in the CABRI in-pile tests [238, 239]. The CABRI-2 tests, for example, focused on transient fission gas behaviour under various heating rates and cladding temperature conditions. Together with the results from many other existing in-pile and out-of-pile tests summarized in Ref. [239], it led to an improved understanding of transient fuel performance and grain boundary behaviour in particular and consequently helped in the development of improved fission gas models.

10.7. FUTURE TRENDS AND POSSIBILITIES

Following the demise in the last decade of important fast reactor programmes in France, Germany, Italy, the UK and, to an extent, Japan, the first step towards the realization of a long term sustainable nuclear programme will be to reverse this trend [240] and to begin to deploy the current designs of fast reactors using MOX fuels with up to 30% fissile plutonium. The previous development programmes, with large demonstration plants having been built and operated, have achieved a high level of knowledge which can provide a solid foundation for the initial installation of full size plants.

In Japan the experimental fast reactor Joyo is under reconstruction to upgrade to the Mk-III core to improve irradiation performance. The first criticality of the Mk-III core is expected in 2002.

Most of the efforts on FBR MOX fuel developments concentrate on improvement to economy and burnup extension. Economy of the FBR system is closely related to fuel design. Burnup extension is one of the most effective approaches for cost reduction. Although burnup extension generally implies a reduction in the fuel cycle cost and an improvement in the availability factor of the reactor plant, it tends to increase the plant capital cost. This is especially true if the mismatch in specific power between fresh fuel and high burnup fuel is increased due to burnup extension. It is therefore important to optimize fuel design with burnup extension in order to reduce the total cost (including the capital cost of the reactor plant).

Fuel pin diameter should be looked at first when considering core optimization because it dominates most of the other design parameters including fissile content, breeding ratio, specific power, breeding gains and fissile specific inventory. If the design requirement on the doubling time is relaxed, or replaced by a requirement on breeding ratio, there is no need to keep the fuel pin outer diameter as small as it is to have high specific power and low fissile specific inventory. From the economic point of view, a large diameter fuel pin has many advantages because it reduces fuel fabrication cost per unit weight of fuel and because it is effective in lowering any power mismatch due to the burnup extension. It is assumed that the economically optimum outer diameter of high burnup fuel is much greater than those designed in the past. Design studies from this viewpoint are being conducted on high burnup cores with large diameter fuel pins [241].

The fuel pin cladding material is clearly a key aspect in the design of future fast reactors. Stainless steel (AISI 316) has been the reference material for many years. However, the neutron induced void swelling of this material can be very large. Development of materials has concentrated on those with better resistance to swelling and irradiation creep [235]. When considering the neutron exposures experienced by fast reactor materials, oxide dispersion strengthened ferritic/martensitic steel is seen as a candidate material for cladding within future fast reactors. The material has the excellent swelling resistance of any ferritic steel, while the high temperature creep rupture strength is increased by the oxide dispersion strengthening technique [240].

While a number of countries have reduced or postponed their fast reactor development programmes, other countries, Japan in particular, continue with new work programmes. Examples include:

(a) A development programme on a simplified and shortened pellet fabrication process, the ‘short process’, is under way at JNC, the aim of which is to reduce the cost of MOX fuel fabrication markedly [42].

(b) Long life materials for structural components are essential for extending the fuel discharge burnup. Oxide dispersion strengthened steels are being developed to endure a fast neutron exposure of 250 dpa at 700°C [242].

(c) Feasibility studies on a commercial FBR cycle system are also being conducted in Japan. In addition to sodium cooled and MOX fuelled FBR systems, various reactor and cycle options and various types of fuel material are being investigated as candidate options for the future [243].

(d) In the Russian Federation construction has started of the BN-800 reactor in which it is planned to use MOX fuel.

Reprocessing of fast reactor fuels has also been studied experimentally and some experience gained. To improve the breeding ratio of the current plant designs, new core layouts have been analysed and, more importantly, advanced fuel materials based on carbide and nitride matrices tested on an experimental scale. Other advanced FBR fuel concepts have concentrated on the more advanced fabrication methods using gelation, producing the sphere-pac and vibro-packed fuel concepts as well as traditional pellets. Such fabrication concepts can be linked to advanced reprocessing flow sheets which lead directly to the co-production of the recycled fuel in one centralized plant. In the Russian Federation, for example, it is intended to recycle plutonium in commercial FBRs and there are plans to construct three such reactors to make more efficient use of the plutonium.

The reactor development work performed until recently was mainly based on sodium cooled fast reactors. Increased safety is claimed for new designs using lead/bismuth eutectics as coolant (the addition of bismuth mitigates the corrosion issues surrounding the use of lead coolant) and the earlier studies on gas cooled fast reactors are also receiving renewed attention. All of these concepts will need further evaluation when considered not only as breeders but also as platforms for inert matrix fuels and for the incineration of plutonium and higher actinides, which are a burden for waste repositories.

Homogeneous and heterogeneous options exist for possible recycling of minor actinides in fast reactors. Homogeneous burning implies minor actinides mixed with MOX fuel in a conventional core (with minor actinide loadings of a few per cent), while heterogeneously burning minor actinides mean the use of dedicated ‘target’ subassemblies fuelled with minor actinides sited in and around a conventional MOX fuelled core. An advantage of the heterogeneous option is the minimal increase in the sodium void reactivity. There are a number of technical and economic issues to be examined when considering the inclusion of minor actinides in a fast reactor core, such as the availability of industrial scale technology for fuel manufacture and reprocessing of fuels containing minor actinides. Additional shielding requirements are needed for handling americium and curium, for instance, and dedicated minor actinide burning reactors are therefore an option to be considered.

Any large scale introduction of fast reactors, as suggested here, is dependent on having designs which, from the outset, are economic under the then prevailing conditions and which are sure of positive acceptance by the public. An incident-free development phase with the commercial plants is an essential pre-condition for the large scale use of fast reactors.

10.8. CONCLUSIONS

Experimental results from fast reactor systems with MOX fuel indicate that future large scale FBRs can be achieved with this fuel type. In addition, a programme of FBRs could contribute to a reduction of transuranic materials because of the flexibility of the design in accommodating different fuel types and compositions. A combination of FBRs and existing LWRs may provide a more favourable and flexible system from the economic and environmental points of view. However, cost efficiency and public acceptance are essential preconditions for the future utilization of fast reactors as either breeders or burners.

11. ALTERNATIVE APPROACHES

11.1. INTRODUCTION

In the opening section, three developments were identified which could protect plutonium from unauthorized diversion, while at the same time allowing the sustainable use of nuclear power and the full realization of the energy potential of natural uranium using plutonium fuel into the distant future:

(a) Repeated recycle of uranium–plutonium fuels in reactor cores,

(b) Development of an advanced and effective plutonium (and actinide) nuclear incineration system to destroy plutonium by conversion to fission product wastes,

(c) Development of inert matrix (uranium-free) fuels using plutonium as the fissile material to minimize the production of fresh plutonium by avoiding neutron capture in a uranium matrix.

Each of these developments is independent but can also be used in combination with one or all of the others depending on the status of nuclear power development at the time.

For (a), the classical fast reactor will be needed, installed in considerable numbers to master the still growing stockpiles of separated plutonium. These power producing plants can be optimized for maximum breeding or minimum fissile production (conversion) but also have the important characteristic of allowing multiple recycle of the plutonium. An essential component of this system is a series of closely coupled reprocessing or ‘fissile extraction’ process plants meeting all the requirements for a safe and emission-free facility. The final output of the exhausted plutonium material could be cycled to the next development, the incineration facility.

Development (b) involves the deployment of several plutonium/minor actinide (MA) incineration facilities where, by means of neutron bombardment, the surplus plutonium/MA may be transmuted into non-fissile radioactive waste. Since more than one pass may be necessary, this development also relies on an effective but safe reprocessing of the intermediate products. This system will always be needed at the end of a nuclear era to destroy finally any remaining plutonium and avoid the safeguards problem of its long term storage in a waste repository.

For (c), since plutonium is inevitably produced from uranium-based fuel as a result of the irradiation of the fertile ^{238}U fuel matrix, attention is being given to developing a uranium-free fuel, either so-called inert matrix or thorium based fuel. The use of an inert matrix fuel (IMF), which has plutonium as the fissile component, would greatly reduce the formation of further plutonium and aid the management (reduction) of separated plutonium inventories. It, also, can be combined with (b) to produce finally a largely plutonium-free waste end product suitable for final geological storage. The advantages of using thorium are the existing fuel technology experience and its applicability in LWRs without changes to the safety systems. The shorter term availability makes it especially attractive for the reduction of weapons grade material stockpiles. The characteristics of spent thorium/plutonium fuel also contribute to enhanced proliferation resistance and therefore make it suitable for longer term interim storage or direct disposal.

11.2. ADVANCED FAST BREEDER AND CONVERTER REACTORS

For the realization of a long term, sustainable, nuclear power programme able to deal with the production and management of plutonium, the first stage will be to begin the deployment of fast reactor systems. Reprocessing of fast reactor fuels belongs to this strategy, building on the small amount of experience already gained. To this must be added advanced methods of MOX fuel fabrication. With the experience and demonstration of sodium cooled fast reactors on an industrial scale, parallel work must be undertaken on new designs using, say, lead/bismuth as coolant or even a return to the gas cooled fast reactor concept investigated in the 1970s and 1980s [244, 245].

An intermediate, but innovative, concept, which may be an opening to the wider deployment of small economic reactor plants, is the South African project for a pebble bed modular reactor (PBMR) [246], and/or the gas turbine modular helium cooled reactor (GTMHR) [247]. For the purposes of this discussion, these HTRs are converter reactors with high intrinsic safety features, which can accept a range of fuel materials (uranium, thorium–^{233}U and also uranium–plutonium). The key feature here is that very high fuel burnups can be achieved with a fuel form (coated particle fuels embedded in a graphite matrix) that makes it difficult to reprocess but which has good characteristics for final direct disposal of the spent fuel (good safeguardability). Such a reactor design with its helium cooling and inert fuel form could become the workhorse of a future thermal reactor programme in the long term.

The use of 100% MOX cores in LWRs is also being studied (Section 4), which would allow more plutonium to be under irradiation and also possibly allow a more effective and earlier introduction of IMFs.

The CANDU heavy water reactor concept has also been studied for the burning of plutonium bearing fuels. The MOX fuel concept uses a standard CANDU 37 pin bundle in which the outer ring contains MOX fuel pellets (1.4% plutonium) with a depleted uranium matrix, the second ring has fuel with 2.3% plutonium, and the inner region has depleted uranium and 5% dysprosium oxide acting as a burnable poison. The destruction rate of the plutonium is estimated to be 1.7 t per GW(e) installed capacity, giving a reduction of plutonium levels of 60% in the fuel bundle on discharge. The MOX fuel bundle is compatible in terms of performance and safety aspects with the normal CANDU core [248].

The heavy water reactor design known as ATR Fugen (165 MW(e)) has also utilized MOX fuels. The fuel design consisted of bundles of 28 fuel pins comprising three layers having 1.8–2.3% plutonium in a natural uranium matrix. So far, 130 t of MOX (1.7 t plutonium) have been loaded since operation commenced in 1979, with MOX core loading rates of between 34 and 72%. A 100% MOX core is also possible. A feasibility study on the disposition of weapons plutonium in ATR having a 100% MOX core with the standard 28 pin fuel bundle indicated a potential disposition rate of 170 kg/a with a net plutonium destruction efficiency of 30% — corresponding figures of 210 kg/a and 42% were also shown to be possible with a 54 pin bundle. In spite of the experience of Fugen, construction of an ATR commercial plant was abandoned for economic reasons in 1995, and the operation of Fugen will also be terminated in 2003.

11.3. PLUTONIUM AND ACTINIDE PARTITIONING AND TRANSMUTATION

Efforts have been underway for a number of years to study and evaluate the separation of plutonium and MAs out of waste streams which are intended for waste repositories containing only irradiated structural materials and radioactive fission products. This is largely to reduce the long term toxic and radiation hazards in such repositories and to reduce the costs by simplifying the repository design requirements. Once separated, the plutonium and MA are to be transmuted into less harmful and shorter lived fission products. Neutron bombardment of the target nuclides requires the development of suitable target materials and designs and the associated irradiation facility, either reactor based or using particle accelerators, called accelerator driven systems (ADSs).

Perhaps the most comprehensive programme now under way and involving all of these aspects is being run in France where the CEA and the Agence nationale pour la gestion des déchets radioactifs (ANDRA) have responsibility to the government authorities to develop a partitioning and transmutation (P&T) system on a defined timescale. Various European partners support the CEA work. Other programmes are supported by the European Community Joint Research Centre and the OECD NEA, in addition to a number of national projects also in place.

In France, one of the first aims of the national waste programme is to reduce the radiotoxicity of nuclear waste through transmutation of the most harmful long lived radionuclides. Both MAs and certain long lived fission products (LLFPs) have a large influence on the long term safety of geological repositories. The scientific feasibility of several advanced separation processes for uranium, plutonium, MAs and LLFPs is stated to be within reach.

For the transmutation stage, the need to optimize target designs is recognized and two paths are being examined. The first is based on the current thermal reactor designs with new fuels to permit the multiple recycling of plutonium and MAs. The second line of study covers the use of fast reactor systems or innovative hybrid solutions coupled to accelerator driven subcritical systems. Complete systems of transmutation are also being examined using, for example, molten salt reactors and pyrochemical systems of fuel reprocessing.

In spite of considerable progress in this area, the French view remains that the partitioning and transmutation approach will not be able to resolve all problems but may just reduce their acuteness.

ADSs are moving from the stage of paper evaluations and physics studies to testing and demonstration of individual components and concepts. The aim is to drive a subcritical arrangement of plutonium and MA materials using a beam of subatomic particles to generate an intensive flux of neutrons causing transmutation to stable or short lived nuclides, which can be efficiently disposed of. Separation of the key elements calls for an advanced reprocessing regime.

One such ADS scheme is the so-called Energy Amplifier proposed by CERN in Geneva, which is a fast neutron subcritical system driven by a proton accelerator. Passive safety is claimed but a large number of physical and safety questions still need to be answered. The source of neutrons is a spallation neutron source in the form of a liquid metal target of lead/bismuth. Such a target concept is under study in Switzerland. The Energy Amplifier is part of a major European collaboration with the R&D programme, funded partly by the fifth framework programme of the European Union, aimed at producing a prototype system [2].

11.4. ADVANCED FISSILE MATERIAL EXTRACTION (REPROCESSING)

The evolutionary improvements to aqueous (Purex) reprocessing of spent reactor fuel, which are surely possible, represent only small steps on the way to achieving a simple robust and emission-free process for the extraction of unused uranium and plutonium. Pyroelectrochemical processes represent a more revolutionary approach. These are dry processes for which the following advantages are claimed:

- High chemical stability of the medium;
- High concentration of the fission elements;
- Batch processing in a single process vessel;
- Ability to handle all forms of fuel, metal, oxides, nitrides, etc.;
- Lack of neutron moderators and therefore increased criticality safety;
- Minimized volume of highly active wastes;
- Recycled fuel product in granulate form which may be loaded directly into fuel rods (vibro-packing).

These systems are under development in France, Japan, the Russian Federation, the USA and elsewhere. In the Russian Federation a semi-industrial plant has been constructed for reprocessing, waste treatment and fuel fabrication, and the resulting Vipac fuels have been widely tested in various fast reactor configurations. On the basis of the concept of a ‘nuclear island’, with one small reprocessing facility and two fast reactors, the concept is claimed to be economic when compared with a single WWER plant. High intrinsic safety features are also claimed [249].

For nitride fuels (Section 11.5.3), a Purex method may be used for reprocessing, but in Japan an innovative pyrochemical reprocessing method has been developed. This consists of a self-completed nitride cycle using molten LiCl–KCl salts, electro-refining and a fabrication process of sphere-pac fuel. An experiment to investigate electro-refining actinide nitrides such as PuN is being developed by JAERI [250].

11.5. ALTERNATIVE FUELS

A principal aim of the development of alternative fuel assembly and reactor designs is proliferation resistance of the plutonium cycle and/or an improved plutonium reduction rate. For both IMFs and thorium based fuels, conceptual studies have been performed. These studies came to the conclusion that the use of these types of fuels in LWRs may be possible. Further development is required.

11.5.1. Inert matrix fuels

Classical nuclear fuels are oxides of uranium in which a buildup of plutonium occurs due to neutron capture and decay during irradiation. A large proportion of this plutonium itself contributes to the fission energy production during irradiation in reactors but, at the end of its life, the fuel still consists of small amounts of fissile uranium and plutonium. This forms the utility’s growing inventory of plutonium, which may either be kept in the spent fuel or be separated by reprocessing for storage or reuse. To avoid this inventory buildup, uranium-free fuels known as IMFs are being studied. Here, the fissile material is plutonium which itself becomes exhausted or burnt out during irradiation but which is not compensated for by the further growth of plutonium from uranium.

The fuel matrix being studied for LWRs is zirconium oxide with the addition of some 5% of fissile plutonium. For reasons of neutronics, the absorption cross-section of the fuel is enhanced to the level of standard uranium fuel by the addition of erbium. Some fuel pellets made of the pressed and sintered ceramic $(Zr_{0.74}Y_{0.14}Er_{0.4}\,Pu_{0.8})O_2$, produced by various fabrication techniques, are being tested, for example, in the HBWR. Further plans are under discussion for the selection of the optimal characteristics of such fuel to be tested in a commercial LWR. Other fuels, for example based on the spinel family, are also being examined. Clearly the optimum fuel ceramic structure has not yet been found [3, 251].

11.5.2. Thorium fuels

In a similar way to IMF, thorium as a carrier material provides a significant potential for effective plutonium management. Thorium fuel has been investigated for some time and found to be suitable for use as a recycling fuel, either with uranium based fuel or in ‘all thorium’ cores. For LWRs, the once through fuel cycle with plutonium as the initial fissile material has the potential for significantly higher plutonium reduction and degradation rates (since no new plutonium is produced) than uranium/plutonium fuel.

Some earlier experience exists with the operational behaviour of thorium/plutonium fuel in LWRs [252], but limited to low plutonium contents. Extending the qualification basis is required before the commercial utilization of thorium technology can be established. However, utilization of the entire potential of the thorium resources requires advanced fuel cycle strategies, together with new developments in technology.

11.5.3. Carbide and nitride fuels

Mixed uranium–plutonium monocarbide and mononitride fuels are considered advanced FBR fuels on the basis of their higher breeding ratio and shorter doubling time, better thermal conductivity and excellent compatibility with sodium coolant and stainless steel cladding. Fuels of this type with 20–30% plutonium have demonstrated satisfactory performance to very high burnups in various sodium cooled test reactors. In India, a hitherto untried mixed uranium–plutonium carbide with a plutonium-rich composition $(Pu_{0.7}U_{0.3})C$ as driver fuel has been utilized in the Fast Breeder Test Reactor (FBTR). A burnup of 72 000 MW·d/(t HM) has been achieved without failure with low fuel swelling and low gas release. Recently, the licensed burnup limit has been increased to 100 000 MW·d/(t HM). Studies are continuing on fuels and materials in support of the Indian Prototype Fast Breeder Reactor (PFBR) [253].

For fast reactors, including studies on the incineration of plutonium and actinides, a ceramic fuel based on a plutonium–zirconium nitride inert matrix is one of those being studied. Fabrication and characterization trials are under way and are to be followed by a test programme in the Phénix fast reactor in France.

Nitrides are produced by carbothermic reduction of oxides under nitrogen. They are chemically stable and can also be adapted for production by sol-gel methods. Tests to date have shown, in FBR conditions, low fission gas release and no pellet–cladding interaction up to at least 50 GW·d/kg burnup.

The high density of the fuel makes it attractive also as a mixed nitride for transmutation concepts. A helium cooled fast reactor concept is under study by JAERI for burning actinides. An actinide kernel with a composition of (66% NpAmCm–34% Pu)N is coated with TiN to form a fuel particle about 1.5 mm in diameter. The particles are held between two porous frits through which helium flows as coolant allowing very high power densities to be obtained due to the efficient cooling. Concept studies are underway.

A further use of nitrides is in the gas cooled HTGR pebble bed concept where the fissile fuel kernels are plutonium nitride allowing almost complete plutonium removal. The long term stability of the graphite fuel balls suggests that the spent fuel can be directly brought into a repository avoiding any reprocessing.

11.5.4. Rock-like fuels

The IMF based on nitrides described above is in the class of ceramic materials known as rock-like fuels. These are tailor made multiphase fuels consisting of mineral like compounds that are chemically and thermodynamically so stable that they are not soluble in nitric acid in normal ways. During irradiation, the solid fission products are precipitated as new mineral like compounds in the fuels making them stable for geological time periods. Accordingly, they are suitable candidates for direct disposal without further reprocessing.

11.6. CONCLUSIONS

In summary, to manage the plutonium produced from an ongoing and sustainable long term nuclear power programme worldwide, the following facilities will be needed. Most of them are still to be realized, developed and/or deployed:

(a) A viable economic nuclear power system containing the items below, which includes a robust international transport regime with high public acceptance;
(b) A system of fast (breeder) and advanced converter reactors as power producers;
(c) A system of strategically located reprocessing (or fissile extraction) plants coupled with a fuel fabrication facility and with small environmental impact;
(d) A number of incineration facilities for final destruction of surplus plutonium;
(e) Uranium-free nuclear fuel materials for use in thermal or fast reactors with characteristics of stability in long term geological storage (the once through cycle);
(f) The necessary waste disposal repositories (plutonium-free to avoid long term safeguards obligations), preferably internationally owned and operated.

The central aspect for plutonium management is the protection of plutonium from unauthorized diversion while at the same time allowing the sustainable use of nuclear power. For the realization of a long term nuclear power programme able to deal with the production and management of plutonium, an important step will be the deployment of fast reactor systems including advanced reprocessing and MOX fuel fabrication methods.

An attractive intermediate concept is the gas cooled high temperature reactor system, operated with IMF forms. Uranium-free fuel types for use in LWRs offer the potential for the reduction of existing plutonium inventories.

In addition, systems will be needed to destroy any remaining plutonium and so to avoid the long term safeguards obligations. Systems for the separation of plutonium and minor actinides from the waste and their transmutation into non-fissile radioactive waste can lead to that goal being achieved. These advanced concepts, and their intrinsically safe design, are a prerequisite for a high degree of public acceptance and the future of nuclear power.

REFERENCES

[1] PORTA, J., BERGERON, J., VASILE, A., LO-PINTO, P., "Transmutation of plutonium, minor actinides and long lived fission products: from LWR to FR", Partitioning and Transmutation (Proc. Swiss Nucl. Soc. Sem. Zurich, 2000), (DEGUELDRE, C., FERRONI, F., Eds), ETH, Zurich (2000).

[2] KADI, Y., "Transmutation potential of the energy amplifier demonstration facility", ibid.

[3] LEDERGERBER, G., DEGUELDRE, C., HEIMGARTNER, P., POUCHON, M.A., KASEMEYER, U., "Inert matrices for transmutation and utilisation of plutonium", Plutonium 2000 — International Conference on the Future of Plutonium (Proc. Conf. Brussels, 2000), Belgian Nuclear Society, Brussels (2000) 28.

[4] FUKUDA, K., et al., "MOX fuel use as a back-end option: trends, main issues and impacts on fuel cycle management", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 6–25 (CD-ROM).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Communication Received From Certain Member States Concerning Their Policies Regarding the Management of Plutonium, INFCIRC/549, IAEA, Vienna (1998).

[6] LECLERE, J., et al., "MOX fuel fabrication and utilisation in fast reactors worldwide", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 49–73 (CD-ROM).

[7] OECD NUCLEAR ENERGY AGENCY, Management of Separated Plutonium — The Technical Options, OECD NEA, Paris (1997).

[8] STRATTON, R.W., "Closing statement", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 531–534 (CD-ROM).

[9] US DEPARTMENT OF ENERGY, Assessment of Plutonium Storage Safety Issues at Department of Energy Facilities, Rep. DOE/DP-0123T, USDOE, Washington, DC (1994).

[10] NATIONAL ACADEMY OF SCIENCES, Management and Disposition of Excess Weapons Plutonium — Reactor Related Options, National Academies Press, Washington, DC (1995).

[11] WILSON, P.D. (Ed), The Nuclear Fuel Cycle — From Ore to Waste, Oxford University Press, Oxford (1996).

[12] KOIZUMI, M., OHTSUKA, K., ISAGAWA, H., AKIYAMA, H., TODOKORO, A., Development of a process for the co-conversion of Pu–U nitrate mixed solutions to mixed oxide powder using a microwave heating method, Nucl. Technol. **61** (1983) 55–70.

[13] ZAKHARKIN, B.S., et al., "Further developments of 'GRANAT' technology of granulated U–Pu fuel", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 146–149 (CD-ROM).

[14] KINCAID, L., AITKEN, E., TAYLOR, I., COPRECAL — co-conversion of U/Pu mixed nitrate to mixed oxide, Trans. Am. Nucl. Soc. **33** (1979) 470.

[15] SNEIDER, V., HERMANN, F., DRUCKENBRODT, W., The AU/PuC process. A co-precipitation process with good product homogeneity to the full scale of plutonium concentration, Trans. Am. Nucl. Soc. **31** (1979) 176.

[16] HAAS, D., SOMERS, J., WALKER, C., BREMIER, S., "Advanced MOX fabrication methods for LWRs", Plutonium Futures — The Science (Proc. Conf. Santa Fe, 2000), (PILLAY, K.K.S., KIM, K.C., Eds), American Institue of Physics, New York (2000).

[17] HAAS, D., SOMERS, J., CHAROLLAIS, F., FUCHS, C., FOURCAUDOT, S., "Fabrication and characterization of MOX fuels with high plutonium content using alternative processes", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 523–530 (CD-ROM).

[18] WICK, O.J. (Ed.), Plutonium Handbook — A Guide to the Technology, 2 Vols, American Nuclear Society, La Grange Park, IL (1980).

[19] Belgian tests easing gallium out of MOX, Nucl. Europe Worldscan, Vols 1 and 2, Jan–Feb (1998) 68.

[20] GREENE, S., "Reactor-based plutonium disposition: opportunities, options and issues", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 478–485 (CD-ROM).

[21] BIBILASHVILI, Y., et al., "Some aspects of a technology of processing weapons grade plutonium to nuclear fuel", ibid., pp. 135–145.

[22] USTINOV, O.A., "Reprocessing of mixed U–Pu fuel by recrystallisation in molten molybdates," ibid., pp. 520–522.

[23] GLAGOVSKI, E., et al., "CHEMOX: an integrated facility for the conversion of Russian weapon-grade plutonium into oxide for MOX fuel fabrication", Plutonium Futures — The Science (Proc. Conf. Santa Fe, 2000), (PILLAY, K.K.S., KIM, K.C., Eds), American Institue of Physics, New York (2000) 214.

[24] BAIRIOT, H., et al., “Overview of MOX fuel fabrication activities”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 81–101 (CD-ROM).

[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Safe Handling and Storage of Plutonium, Safety Reports Series No. 9, IAEA, Vienna (1998).

[26] NADYKTO, B.A., TIMOFEEVA, L.F., “New nuclear safe plutonium ceramic compositions with neutron poisons for plutonium storage”, Plutonium Futures — The Science (Proc. Conf. Santa Fe, 2000), (PILLAY, K.K.S., KIM, K.C., Eds), American Institute of Physics, New York (2000) 73–74.

[27] LEIGH, B.D., “Plutonium dioxide — a managed resource for the present and the future”, Global 1995 — Evaluation of Emerging Nuclear Fuel Cycle Systems (Proc. Int. Conf. Versailles, 1995), Vol. 1, American Nuclear Society, La Grange Park, IL (1995).

[28] BAIRIOT, H., “MOX fuel: the accomplishment and the future”, TOPFUEL ’97 (Proc. Conf. Manchester, 1997), Vol. 1, BNES, London (1997) 4.50–4.68.

[29] BAIRIOT, H., et al., “Overview of MOX fuel fabrication achievements”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 81–101 (CD-ROM).

[30] INTERNATIONAL ATOMIC ENERGY AGENCY, Environmental Aspects Based on Operational Performance of Nuclear Fuel Fabrication Facilities, IAEA-TECDOC-1306, Vienna (2002).

[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Advanced Methods of Process/Quality Control in Nuclear Reactor Fuel Manufacture (Proc. Tech. Comm. Mtg Lingen, 1999), IAEA-TECDOC-1166, Vienna (1999).

[32] HUGELMANN, D., GRENECHE, D., “MELOX fuel fabrication plant: operational feedback and future prospects”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 102–108 (CD-ROM).

[33] OKITA, T., AONO, S., ASAKURA, K., AOKI, Y., OHTANI, T., “Operational experiences in MOX fuel fabrication for the Fugen advanced thermal reactor”, ibid., pp. 109–117.

[34] DERAMAIX, P., EECKHOUT, F., PAY, A., PELCKMANS, E., “Experience and trends at the Belgonucléaire plant”, ibid., pp. 169–176.

[35] VAN VLIET, J., et al., “MOX fuel fabrication, in-reactor performance and improvement”, ENC ’98 (Proc. Int. Conf. Nice, 1998), European Nuclear Society, Bern (1998) 233–242.

[36] EDWARDS, J., BRENNAN, J.M., MACLEOD, H.M., BROWN, C., MOX fuel at BNFL, Nucl. Eng. **37** 6 (1996) 178–181.

[37] KRELLMANN, J., Plutonium processing at the Siemens/Hanau fuel fabrication plant, Nucl. Technol. **102** (1993) 18–28.

[38] KAMATH, H.S., ANANTHARAMAN, K., PURUSHOTHAM, D.S.C., “MOX fuel for the Indian nuclear power programme”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 190–199 (CD-ROM).

[39] MAYORSHIN, A.A., et al., “Validity of using $UPuO_2$ vibropack experimental fuel pins in reactors on fast and thermal neutrons: First experiments on conversion of weapons grade plutonium into nuclear fuel”, ibid., pp. 150–159.

[40] NUCLEAR FUEL, Europeans first in line for SMP; no fuel for Japan until about 2004, Nucl. Fuel, April (2000).

[41] KRELLMANN, J., “Cogema Cadarache”, Jahrestagung Kerntechnik, May (1998) 347–351.

[42] ASAKURA, K., AONO, S., YAMAGUCHI, T., DEGUCHI, M., “Current developments of fuel fabrication technologies at the plutonium fuel production facility PFPF”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 118–126 (CD-ROM).

[43] IVANOV, V., “Russian program for plutonium involved in the fuel cycle of NPP”, TOPFUEL 2001 (Proc. Int. Conf. Stockholm, 2001) (2001) paper 1–6 (CD-ROM).

[44] MORKOVNIKOV, V., et al., “Continuous process production for MOX fuel fabrication according to GRANAT technology”, Plutonium Futures — The Science (Proc. Conf. Santa Fe, 2000), (PILLAY, K.K.S., KIM, K.C., Eds), American Institute of Physics, New York (2000) 199.

[45] VARYKHANOV, V., “Comparative analysis of basic process arrangements for converting surplus weapons grade plutonium to MOX fuel”, ibid., p. 208.

[46] THE NUCLEAR INSTALLATIONS INSPECTORATE OF THE HEALTH AND SAFETY EXECUTIVE, An Investigation into the Falsification of Pellet Diameter Data in the MOX Demonstration Facility at the BNFL Sellafield Site and the Effect of this on the Safety of MOX Fuel in Use, HSE, London (2000).

[47] NUCLEAR FUEL, German regulators go to Cogema plant to check QC flaws with MOX at Isar-2, Nucl. Fuel **26** (April 2000) 7.

[48] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidebook on Quality Control of Mixed Oxides and Gadolinium Bearing Fuels for Light Water Reactors, IAEA-TECDOC-584, Vienna (1991).

[49] FRESHLEY, M., et al., Nucl. Technol. **15** (1972) 239.

[50] VANDEZANDE, J., “Microscopic determination of the PuO_2 grain size and pore size distribution of MOX pellets with an image analysis system”, Advanced Methods of Process/Quality Control in Nuclear Reactor Fuel Manufacture (Proc. Tech. Comm. Mtg Lingen, 1999), IAEA-TECDOC-1166, IAEA, Vienna (1999) 59–64.

[51] GARCIA, P., BOULORE, A., GUERIN, Y., TROTABAS, M., GOEURIOT, P., “In-pile densification of MOX fuels in relation to their initial microstructure”, Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Park City, Utah, 2000), American Nuclear Society, La Grange Park, IL (2000) 679–693 (CD-ROM).

[52] GUERIN, Y., et al., “Microstructure evolution and in-reactor behaviour of MOX fuel”, ibid., pp. 706–719.

[53] IVISON, P.K., COOK, P.M.A., BREMIER, S., WALKER, C.T., “Quantification of the homogeneity of BNFL SBR MOX fuel using compositional X ray mapping”, Nuclear Fuel Behaviour Modelling at High Burnup and its Experimental Support (Proc. IAEA Tech. Comm. Mtg Windermere, 2000), IAEA-TECDOC-1233, IAEA, Vienna (2001) 239–246.

[54] LEE, B.-H., KOO, Y.-H., SOHN, D.-S., “Modelling of MOX fuel’s thermal conductivity considering its microstructural heterogeneity”, ibid., pp. 247–256.

[55] HESKETH, K.W., THOMAS, G.M., ROBBINS, C., “Elimination of the homogenisation step in the manufacture of MOX assemblies for LWRs by means of a reactivity equivalence formulation”, Jahrestagung Kerntechnik ’92 (Proc. Conf. Karlsruhe, 1992), INFORUM, Bonn (1992) 225–228.

[56] CROUSILLES, M., Mesures et systèmes d’informations pour la gestion des matières nucléaires dans l’usine de fabrication de combustibles MOX de Cogema Cadarache, RGN **1** (2001) 37–40.

[57] EDWARDS, J., BROWN, C., MARSHALL, S., CONNELL, M., THOMPSON, H., “The development of BNFL’s MOX fuel supply business”, RECOD ’98 (Proc. 5th Int. Conf. Nice, 1998), Vol. 1, European Nuclear Society, Bern (1998) 182.

[58] COOCH, J.P., “Meeting customer requirements in the Sellafield MOX Plant”, TOPFUEL ’01 (Proc. Int. Conf. Stockholm, 2001), Swedish Nuclear Society, Stockholm (2001) paper 2B-3 (CD-ROM).

[59] “Stage ’97”, PNC’s Year in Review, PNC report, Japan (1997).

[60] NUCLEAR FUEL, Japanese worry how to minimize Pu surplus when Rokkasho runs, Nucl. Fuel, May (2000) 9–10.

[61] WARIN, D., FRAIZE, G., “Fabrication MOX — Les procédés envisagés et retenus — Les projets en cours”, Le recyclage civil des matières militaires, Réunion SFEN, Société française de l’énergie nucléaire, Paris (2000).

[62] TSYKANOV, V.A., et al., “Main trends and content of works on fabrication of fuel rods with MOX fuel for the WWER-1000 reactor”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 489–499 (CD-ROM).

[63] Brown, C., CALLENS, C., GOLL, W., LIPPENS, M., “Overview on MOX fuel for LWRs: Design, performance and testing”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 203–212 (CD-ROM).

[64] FUJISHIRO, T., WEST, J.-P., HEINS, L., JADOT, J.J., “Overview of safety analysis, licensing and experimental background of MOX fuels in LWRs”, ibid., pp. 38–48.

[65] PORSCH, D., CHARLIER, A., MEIER, G., MOUGNIOT, J.C., TSUDA, K., “Overview on neutronic fuel assembly design and in-core fuel management”, ibid., pp. 339–353.

[66] PROVOST, J.-L., “MOX fuel for PWRs; EDF experience and perspective”, Plutonium 2000 — International Conference on the Future of Plutonium (Proc. Int. Conf. Brussels, 2000), Belgian Nuclear Society, Brussels (2000) 16 (CD-ROM).

[67] WORRALL, A., “Effect of plutonium vector on core wide nuclear design parameters”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 372–381 (CD-ROM).

[68] KURITA, I., KIKUCHI, K., ABE, T., “MOX fuel irradiation behaviour in a thermal reactor”, ibid., pp. 244–252.

[69] BLANPAIN, P., CALLENS, C., GOLL, W., “MOX fuel performance and development”, Topfuel 2001, Swedish Nuclear Society, Stockholm (Proc. Conf. Stockholm, 2001), (2001) paper 4A-3 (CD-ROM).

[70] OECD NUCLEAR ENERGY AGENCY, Physics of Plutonium Recycling — Issues and Perspectives, Vol. 1, OECD NEA, Paris (1995).

[71] BROWN, O.C., et al., "Safety analysis for mixed oxide (MOX) fuel in boiling water reactors (BWRs)", Safety of Operating Reactors (Proc. Int. Top. Mtg San Francisco, 1998), American Nuclear Society, La Grange Park, IL (1998) 282.

[72] SCHLOSSER, G.J., KREBS, W.-D., Experience in PWR and BWR mixed oxide fuel management, Nucl. Technol. **102** (1993) 54.

[73] ROME, M., et al., Plutonium reload experience in French pressurised water reactors, Nucl. Technol. **94** (1991) 87.

[74] PORSCH, D., et al., "High burnup MOX in light water reactors", Plutonium 2000 — International Conference on the Future of Plutonium (Proc. Int. Conf. Brussels, 2000), Belgium Nuclear Society, Brussels (2000) (CD-ROM).

[75] VAN DOESBURG, W., MEADER, C., WAND, H., "Licensing of MOX fuel in Switzerland", Safety of Operating Reactors (Proc. Int. Top. Mtg San Francisco, 1998), American Nuclear Society, La Grange Park, IL (1998) 298.

[76] FABER, C.R., SEMMRICH, J., "Safety aspects in recycling plutonium in LWRs — German experience", ibid., pp. 275–281.

[77] GOLDSTEIN, L., MALONE, J., "Safety and licensing of MOX versus UO_2 for BWRs and PWRs: Aspects applicable for civilian and weapons grade Pu", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 399–409 (CD-ROM).

[78] OECD NUCLEAR ENERGY AGENCY, Fuel Safety Criteria Technical Review, Rep. NEA/CSNI/R(99)25, OECD NEA, Paris (2000).

[79] DEPLECH, M., TOMMASI, J., ZAETTA, A., "Weapons grade plutonium disposition in PWR, CANDU and FR", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 410–417 (CD-ROM).

[80] SCHLOSSER, G., PORSCH, D., "Safe utilization of weapons grade plutonium in existing PWRs — supported by German recycling experience", Fuel Storage Technology (Proc. Top. Mtg Seattle, 1996), American Nuclear Society, La Grange Park, IL (1996) 785.

[81] CLARK, R., et al., "MOX fuel irradiation program for disposition of surplus plutonium", ANS 2000 Annual Meeting (Proc. Mtg San Diego, 2000), American Nuclear Society, La Grange Park, IL (2000).

[82] NUMARK, N., MICHEL, A., Plutonium 2000 — International Conference on the Future of Plutonium (Proc. Conf. Brussels, 2000), Belgian Nuclear Society, Brussels (2000) 1–56.

[83] IZUTSU, S., SASAGAWA, M., AOYAMA, M., MARUYAMA, H., SUZUKI, T., "Progress of full MOX core design in ABWR", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 362–371 (CD-ROM).

[84] PROVOST, J.-L., SCHRADER, M., NOMURA, S., "MOX fuel fabrication and utilisation in LWRs worldwide", ibid., pp. 29–37.

[85] MEYER, R., "NRC activities related to high burnup, new cladding types and mixed-oxide fuel", Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Park City, Utah, 2000), American Nuclear Society, La Grange Park, IL (2000) 736–744 (CD-ROM).

[86] BLANPAIN, P., THIBAULT, X., PAGES, J.P., “Recent results from the in reactor MOX fuel performance in France and improvement programme”, Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Portland, 1997), American Nuclear Society, La Grange Park, IL (1997) 39–45 (CD-ROM).

[87] YAMATE, K., ABETA, S., SUZUKI, K., DOI, S., “MOX fuel design and development consideration”, Recycling of Plutonium and Uranium in Water Reactor Fuel (Proc. IAEA Tech. Comm. Mtg Windermere, 1995), IAEA-TECDOC-941, Vienna (1995) 133–144.

[88] LIPPENS, M., MALDAGUE, T., BASSELIER, J., BOULANGER, D., MERTENS, L., “Highlights on R&D work related to the achievement of high burnup with MOX fuel in commercial reactors”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 220–232 (CD-ROM).

[89] DIMAYUGA, F.C., FLOYD, M.R., COX, D.S., “Fabrication and performance testing of CANDU mixed-oxide fuel”, ibid., pp. 292–299.

[90] “US MOX arrives at Chalk River”, Nucl. Eng. Int. (Feb. 2000) 7.

[91] CADELLI, N., LIPPENS, M., “A European investigation of MOX fuel performance in LWRs”, Nuclear Power Performance and Safety (Proc. Int. Conf. Vienna, 1987), Vol. 5, IAEA, Vienna (1988) 439–458.

[92] BRUNEL, L., BLANPAIN, P., CHAIGNE, G., TROTABAS, M., “MOX fuel performance in French PWR reactors: Recent results and improvement programme”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 213–219 (CD-ROM).

[93] CAILLOT, L., DELETTE, G., PIRON, J.P., LEMAIGNAN, C., “Analytical studies of the behaviour of MOX fuel”, Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Portland, 1997), American Nuclear Society, La Grange Park, IL (1997) 62–69.

[94] PARRAT, D., HARRER, A., “Failed high burnup MOX fuel performance: the EDITHMOX-2 analytical irradiation”, Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Park City, Utah, 2000), American Nuclear Society, La Grange Park, IL (2000) 720–734 (CD-ROM).

[95] SCHMITZ, F., PAPIN, J., GONNIER, C., “RIA tests in CABRI with MOX fuel”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 300–310 (CD-ROM).

[96] LEMOINE, F., PAPIN, J., FRIZONNET, J.M., CASALIS, B., RIGAT, H., “The role of grain boundary fission gas in high burnup fuel under reactivity initiated accident conditions”, Fission Gas Behaviour in Water Reactor Fuel (Proc. Sem. Cadarache, 2000), NEA/NSC/DOC(2000)20, OECD Nuclear Energy Agency, Paris (2000).

[97] GOLL, W., et al., Irradiation behaviour of UO_2/PuO_2 fuel in light water reactors, Nucl. Technol. **102** (1993) 29–46.

[98] WALKER, C.T., et al., Effect of inhomogeneity on the level of fission gas and caesium release from OCOM MOX fuel during irradiation, J. Nucl. Mater. **228** (1996) 8–17.

[99] DOI, S., YAMATE, K., “High burnup MOX fuel and fuel rod design improvement”, Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Portland, 1997), American Nuclear Society, La Grange Park, IL (1997) 46–53.

[100] OGATA, K., et al., “BWR fuel performance and recent R&D activities in Japan”, ibid.

[101] OGUMA, M., MOCHIDA, T., NOMATA, T., ASAHI, K., "Technology developments for Japanese BWR MOX fuel utilization", Recycling of Plutonium and Uranium in Water Reactor Fuel (Proc. IAEA TCM Windermere, 1995), IAEA-TECDOC-941, Vienna (1995) 155–170.

[102] INOUE, S., KUWAHARA, H., ABETA, S., OGAWA, S., "The result of PIE on MOX fuel rods irradiated in Mihama unit 1", Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg West Palm Beach, 1994), American Nuclear Society, La Grange Park, IL (1994) 264–272.

[103] WIESENACK, W., McGRATH, M., "Performance of MOX fuel: an overview of the experimental programme of the OECD Halden Reactor Project and review of selected results", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 253–262 (CD-ROM).

[104] BROWN, C., et al., "Performance of BNFL SBR MOX fuel for the next decade", Top Fuel '99 — LWR Fuel Highlights at the Beginning of the Third Millennium (Proc. Int. Top. Mtg Avignon, 1999), Société français d'énergie nucléaire, Paris (1999) 223–234.

[105] GATES, G.A., WHITE, R.J., "Start-of-life gap conductance measurements on SBR mixed oxide fuel", Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Park City, Utah, 2000), American Nuclear Society, La Grange Park, IL (2000) 696–705 (CD-ROM).

[106] COWELL, B.S., MORRIS, R.N., OTT, L.J., "Surplus plutonium mixed oxide fuel irradiation experiment in the Advanced Test Reactor: final post-irradiation results for low-exposure MOX fuel", ibid.

[107] MORRIS, R.N., et al., MOX Average Power Intermediate PIE: 21 GW·d/MT Final Report, ORNL/MD/LTR-99, Oak Ridge Natl Lab., TN (2000).

[108] BLANPAIN, P., BRUNEL, L., THIBAULT, X., TROTABAS, M., "MOX fuel performance in the French PWRs: status and development", Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Park City, Utah, 2000), American Nuclear Society, La Grange Park, IL (2000) 693–705 (CD-ROM).

[109] COOK, P., STRATTON, R., WALKER, C.T., "Post-irradiation examination of BNFL MOX fuel", ibid, pp. 653–663.

[110] COOK, P., PALMER, I., WALKER, C., STRATTON, R., "PIE of BNFL's first commercially irradiated SBR MOX fuel", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 233–243 (CD-ROM).

[111] KREBS, W.D., GOLL, W., "Operational performance of MOX fuel in light water reactors", paper presented at ANS Winter Mtg, Albuquerque, 1997.

[112] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Fuel Behaviour Modelling at High Burnup and its Experimental Support (Proc. Tech. Comm. Mtg Windermere, 2000), IAEA-TECDOC-1233, Vienna (2001).

[113] SONTHEIMER, F., LANDSKRON, H., "Puzzling features of EPMA radial fission gas release profiles: The key to realistic modeling of fission gas release up to ultra high burnup of 100 MW·d/kg M with CARO-E", Nuclear Fuel Behaviour Modelling at High Burnup and its Experimental Support (Proc. IAEA Tech. Comm. Mtg Windermere, 2000), IAEA-TECDOC-1233, IAEA, Vienna (2001) 105–124.

[114] BERNARD, L.C., BLANPAIN, P., "Overall models and experimental database for UO_2 and MOX fuel increasing performance", ibid., pp. 291–304.

[115] ROSSITER, G.D., COOK, P.M.A., WESTON, R., "Isotopic modelling using the ENIGMA-B fuel performance code", ibid., pp. 227–238.

[116] IKEDA, H., KIKUSHI, T., ONO, S., "Improvement of FPAC code", ibid., pp. 277–290.

[117] LIPPENS, M., et al., "Comparative thermal behaviour of MOX and UO_2 fuels", Thermal Performance of High Burnup LWR Fuel (Proc. Sem. Cadarache, 1998), OECD, Paris (1998).

[118] TOPLISS, I.R., PALMER, I.D., ABETA, S., IRISA, Y., YAMATE, K., "Measurement and analysis of MOX physical properties", Recycling of Plutonium and Uranium in Water Reactor Fuel (Proc. IAEA TCM, Newby Bridge, 1995), IAEA-TECDOC-941, IAEA, Vienna (1995) 205–215.

[119] DURIEZ, C., ALESSANDRI, J.P., SERVAIS, T., PHILIPPONNEAU, Y., Thermal conductivity of hypostoichiometric low Pu content $(U, Pu)O_{2-x}$ mixed oxide, J. Nucl. Mater. **277** (2000) 143–158.

[120] ONOUFRIEV, V., STRATTON, R.W., "Status and trends in plutonium recycling in nuclear power reactors", Plutonium and Actinides (Proc. Top. Conf. Santa Fe, 2000), American Institute of Physics, New York (2000) 4–6.

[121] GUERIN, Y., et al., "Microstructure evolution and in-reactor behaviour of MOX fuel", Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg Park City, 2000), American Nuclear Society, La Grange Park, IL (2000) 706–719.

[122] PALMER, I., ROSSITER, G., WHITE R.J., "Development and validation of the ENIGMA code for MOX fuel performance modelling", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 271–281 (CD-ROM).

[123] MORGAN, J.N., KREBS, W.D., "Siemens' experience with large scale mixed oxide fuel assembly insertion in light water reactors", Light Water Reactor Fuel Performance (Proc. ANS Int. Top. Mtg West Palm Beach, 1994), American Nuclear Society, La Grange Park, IL (1994) 711–717.

[124] KAMIMURA, K., KOBAYASHI, Y., NOMATA, T., "Helium generation and release in MOX fuels", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 263–270 (CD-ROM).

[125] STARK, W.A., Helium release from PuO_2 micro spheres, Nucl. Metall. **17** 2 (1970) 554.

[126] GRIMES, R.W., "Simulating the behaviour of inert gases in UO_2", Fundamental Aspects of Inert Gases, Plenum Press, New York and London (1991).

[127] PARRAT, D., LEUTHROT, C., HARRER, A., DANGOULEME, D., "Behaviour of a defective MOX fuel rod in a PWR", Recycling of Plutonium and Uranium in Water Reactor Fuel (Proc. IAEA Tech. Comm. Mtg Windermere, 1995), IAEA-TECDOC-941, IAEA, Vienna (1995) 319–340.

[128] CALLENS, C., PAVAGEAU, O., PROVOST, J.L., THIBAULT, X., "Parity of MOX and UO_2 fuels in EDF 900 MWe PWRs", TOPFUEL '99 — LWR Fuel Highlights at the Beginning of the Third Millennium (Proc. Int. Top. Mtg Avignon, 1999), Société française d'énergie nucléaire, Paris (1999) 203–209.

[129] PIRON, J.P., PELLETIER, M., PAVAGEAU, J., “Helium behaviour in spent UO_2 and MOX fuels”, Fission Gas Behaviour in Water Reactor Fuel (Proc. Sem. Cadarache, 2000), NEA/NSC/DOC(2000)20, OECD Nuclear Energy Agency, Paris (2000).

[130] DYCK, H.P., RAWL, R., VAN DEN DURPEL, L., “The transportation of PuO_2 and MOX fuel and management of irradiated MOX fuel”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 433–445 (CD-ROM).

[131] LANGE, F., MAIRS, J., NIEL, C., “Risk associated with the transport of radioactive materials in the fuel cycle”, Nuclear Fuel Cycle and Reactor Strategies: Adjusting to New Realities (Proc. Int. Symp. Vienna, 1997), IAEA-TECDOC-990, IAEA, Vienna (1997) 161–172.

[132] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, Safety Standards Series No. TS-R-1 (ST-1, Revised), IAEA, Vienna (1996).

[133] UNITED NATIONS, Recommendations on the Transport of Dangerous Goods — Model Regulations, ST/SG/AC.10/1/Rev.12, UN, Geneva (2001).

[134] INTERNATIONAL CIVIL AVIATION ORGANIZATION, Technical Instructions for the Safe Transport of Dangerous Goods by Air, Doc. 9284-AN/905, ICAO, Montreal (2001).

[135] INTERNATIONAL AIR TRANSPORT ASSOCIATION, Dangerous Goods Regulations, IATA Resolution 618, Attachment A, 43rd edn, IATA, Montreal and Geneva (2001).

[136] INTERNATIONAL MARITIME ORGANIZATION, International Maritime Dangerous Goods Code, IMO, London (2000).

[137] ECONOMIC COMMISSION FOR EUROPE, ADR — European Agreement concerning the International Carriage of Dangerous Goods by Road, Rep. ECE/TRANS/140, Vol. 1, UN, Geneva (2001).

[138] RID COMMITTEE OF EXPERTS, Regulations Concerning the International Carriage of Dangerous Goods by Rail (RID), Her Majesty’s Stationary Office, London (2001).

[139] ECONOMIC COMMISSION FOR EUROPE, European Provisions Concerning the International Carriage of Dangerous Goods by Inland Waterway (ADN), UN, Geneva (1997).

[140] INTERNATIONAL MARITIME ORGANIZATION, International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes on Board Ships (INF Code), IMO, London (2000).

[141] INTERNATIONAL ATOMIC ENERGY AGENCY, The Physical Protection of Nuclear Material and Nuclear Facilities, INFCIRC/225/Rev.4, IAEA, Vienna (1999).

[142] INTERNATIONAL ATOMIC ENERGY AGENCY, The Convention on the Physical Protection of Nuclear Material, INFCIRC/274/Rev.1, IAEA, Vienna (1980).

[143] INTERNATIONAL ATOMIC ENERGY AGENCY, Convention on Early Notification of a Nuclear Accident, INFCIRC/335, IAEA, Vienna (1986).

[144] INTERNATIONAL ATOMIC ENERGY AGENCY, Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, INFCIRC/336, IAEA, Vienna (1986).

[145] OECD NUCLEAR ENERGY AGENCY, The Convention on Third Party Liability in the Field of Nuclear Energy, OECD NEA, Paris (1982).

[146] INTERNATIONAL ATOMIC ENERGY AGENCY, International Conventions on Civil Liability for Nuclear Damage, Legal Series No. 4, IAEA, Vienna (1966).

[147] INTERNATIONAL ATOMIC ENERGY AGENCY/ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, Nuclear Liability: Joint Protocol Relating to the Application of the Vienna Convention and the Paris Convention, IAEA/OECD Nuclear Energy Agency, Vienna (1989).

[148] THOMAS, A.B., VERDIER, A., "A review of plutonium transport", paper presented at 32nd Mtg of the ITU Comm., Brussels, 1989.

[149] VERDIER, A., "MOX fuel transport: The French experience", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 446–450 (CD-ROM).

[150] SANCHIS, H., VERDIER, A., "MOX fuel transport: the French experience", TOPFUEL '99 — LWR Fuel Highlights at the Beginning of the Third Millennium (Proc. Int. Top. Mtg Avignon, 1999), Société française d'énergie nucléaire, Paris (1999) 274–281.

[151] ADVISORY COMMITTEE ON THE SAFE TRANSPORT OF RADIOACTIVE MATERIALS, The Transport of Civil Plutonium by Air, HMSO, London (1988).

[152] PILATE, S., et al., Nuclear Data for Advanced MOX Fuels, European Commission/ Nuclear Science and Technology, Rep. EUR 19126 EN, EU, Brussels (2000).

[153] BASSELIER, J., et al., "A detailed investigation of the source term devoted to UO_2 and MOX high burnup LWR fuel through the ARIANE international programme", KTG Fachtagung, Forschungszentrum Karlsruhe (Feb. 1998) 161–170.

[154] BERGELSON, B., et al., "Decay heat power and radiotoxicity of actinides from spent nuclear uranium and uranium–plutonium fuel at long-term storage", Jahrestagung Kerntechnik, JK'2000 (Proc. Conf. Bonn, 2000), INFORUM, Bonn (2000) 275–279.

[155] BELGONUCLEAIRE, STONE, An International Programme on the Characterisation of Stored UO_2 and MOX Nuclear Elements, Rep. ST 99/01 Rev B, Société Belge pour industrie nucléaire, Brussels (2000).

[156] INTERNATIONAL ATOMIC ENERGY AGENCY, Long Term Storage of Spent Nuclear Fuel — Survey and Recommendations, IAEA-TECDOC-1293, Vienna (2000) 49.

[157] GESELLSCHAFT FUR NUKLEAR-BEHALTER, Transport and Storage Cask CASTOR V/52, GNB, Essen (1997).

[158] GRENECHE, D., Cogema, Vélizy, personal communication, 2001.

[159] BOURDET, L., et al., Recyclage du plutonium — études liées au stockage et à la manutention des assemblages combustibles au plutonium, Eur. Appl. Res. Rep., Nucl. Sci. Technol. Sect. **3** 1/2 (1981) 229–244.

[160] RÉPUBLIQUE FRANÇAISE, Loi no 91-1381 du 30 décembre 1991 relative aux recherches sur la gestion des déchets radioactifs, J. Off. Rép. Fr., Lois et Décrets **124** 1 (1992).

[161] NIEZBORALA, M., Stockage et entreposage — les plannings techniques vraisemblables, Journée SFEN "Le cycle et les scénarios" (Jan. 2001) 1–17.

[162] COMMISSION NATIONALE D'ÉVALUATION, Rapport d'évaluation , No. 4 (1998).

[163] CHARPIN, J.-M., et al., Étude économique de la filière électrique nucléaire, Rap. au Premier Ministre, Commissariat Général du Plan, Paris (2000).

[164] BAIRIOT, H., "Pu accounts: technical challenges and industrial solutions", paper presented at IAEA Tech. Comm. Mtg on Perspective of Plutonium Utilisation and Disposition, Brussels, 2000.

[165] PIRON, J.-P., PELLETTER, M., PAVAGEAU, J., "Helium behaviour in spent UO_2 and MOX fuels", Fission Gas Behaviour (Proc. Int. Sem. Cadarache, 2000), CEA, Cadarache (2000).

[166] TSUKADA, T., et al., "Dissolution studies on high burnup UO_2 fuel and MOX LWR fuel", RECOD '98 (Proc. Int. Conf. Nice, 1998), Société française d'énergie nucléaire, Paris (1998) 274–284.

[167] COMMISSARIAT A L'ÉNERGIE ATOMIQUE, ATALANTE 2000 (Proc. Int. Conf. Avignon, 2000), CEA, Paris (2000).

[168] ZAKHARKIN, B.S., "Chemical and technological aspects of spent U–Pu fuel reprocessing", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 451–458 (CD-ROM).

[169] BATAILLE, C., "L'évolution de la recherche sur la gestion des déchets nucléaires à haute activité, tome I: les déchets civils", Assemblée Nationale 2689, Sénat 299, Assemblée Nationale, Paris (1996).

[170] ÉNERPRESSE, UP_2 achève le retraitement du MOX allemand, Énerpresse, Paris (1992).

[171] UNITED REPROCESSORS GESELLSCHAFT, Demonstration of industrial scale MOX fuel reprocessing, Reprocess. News **22** (Nov. 1993) 2–4.

[172] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Facilities other than Reactors, Technical Reports Series No. 386, IAEA, Vienna (1998).

[173] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Fuel Cycle Facilities, Safety Standards Series No. WS-G-2.4, IAEA, Vienna (2001).

[174] REISENWEAVER, D., LARAIA, M., Preparing for the End of the Line, Radioactive Residues from Nuclear Decommissioning, IAEA Bulletin, Vol. 42, No. 3, IAEA, Vienna (2000) 51–54.

[175] HAAS, D., et al., Mixed-oxide fuel fabrication technology and experience at the Belgonucleaire and CFCa plants and further developments for the MELOX plant, Nucl. Technol. **106** (1994) 60–82.

[176] FOURNIER, W., L'expérience de fabrication de combustible MOX et le démarrage de MELOX, Rev. Gén. Nucl. (May–June 1997) 37–39.

[177] IZQUIERDO, J.-J., LAVENU, A., KNIEBIHLI, B., "The centralised alpha waste treatment facility (UCD) at La Hague", RECOD '98: Recycling, Conditioning and Disposal (Proc. Int. Conf. Nice, 1998), Vol. 3, Société française d'énergie nucléaire, Paris (1998) 939–946.

[178] INTERNATIONAL ATOMIC ENERGY AGENCY, Minimization of Waste from Uranium Purification, Enrichment and Fuel Fabrication, IAEA-TECDOC-1115, Vienna (1999).

[179] OECD NUCLEAR ENERGY AGENCY, Decontamination Techniques Used in Decommissioning Activities, A Report by the NEA Task Group on Decontamination, OECD NEA, Paris (1999).

[180] GOLDACKER, H., GALLA, F., KERN, D., Rückbau der Plutoniumextraktionsanlage Pute, Atomwirtschaft **6** (1997) 387–392.

[181] LEDEBRINK, F.W., HAAS, E., “Experience in waste management in a MOX-fuel fabrication plant”, ESARDA (Proc. 14th Mtg Salamanca, 1992), SNS, Madrid (1992) 149–156.

[182] FABER, P., BAUMANN, R., “Shutdown of Siemens MOX-facility in Hanau, the clean out process as first step”, Nuclear Technology (Proc. Ann. Mtg Munich, 1998), INFORUM, Bonn (1998) 533–537.

[183] FABER, P., BAUMANN, R., “Progress at the clean out process of the Siemens MOX facility in Hanau”, Nuclear Technology (Proc. Ann. Mtg Karlsruhe, 1999), INFORUM, Bonn (1999) 511–515.

[184] RUPAR, H., FABER, P., RUHBAUM, M., Der Rückbau des Siemens Brennelementewerks in Hanau und der Heissen Zellen, Atomwirtschaft–Atomtech. **44** (1999) 466–471.

[185] FABER, P., LEDEBRINK, F.W., “Stillegung des Siemens Brennelementewerks Hanau: Endlagergerechte Konditionierung von U- und Pu-haltigen Abfällen und deren Zwischenlagerung am Standort”, Nuclear Technology (Proc. Ann. Mtg Dresden, 2001), Atom-Forum (2001) 287–290.

[186] SCHNEIDER, V.W., LEDEBRINK, F.W., Cementation of TRU waste by a new process: properties of the product, Adv. Ceram. **8** (1984) 394–400.

[187] RUPAR, H., BAUMANN, R., FABER, P., RUHBAUM, M., SCHMITT, H., “Decommissioning of four German fuel cycle facilities”, Waste Management 2000 (Proc. Conf. Tucson, 2000) (2000) 28–40.

[188] LEDEBRINK, F.W., FABER, P., VALENCIA, L., GRAF, A., Conditioning of radioactive waste for long-term interim storage, Atomwirtschaft **45** (2000) 610–614.

[189] FRITZ, P., et al., “Remote dismantling of the spent fuel reprocessing plant Karlsruhe”, KONTEC ’97 Conditioning of Radioactive Operational and Decommissioning Wastes (Proc. Symp. Hamburg, 1997), Kerntechnische Gesellschaft, Bonn (1997).

[190] EIBEN, K., FRITZ, P., “The WAK decommissioning and dismantling program”, SPECTRUM ’96 (Proc. Conf. Seattle, 1996), American Nuclear Society, La Grange Park, IL (1996) 1765–1768.

[191] FOSSATI, G., et al., “The decommissioning of a Pu fuel fabrication pilot facility at the Casaccia Research Centre”, RECOD ’94: Nuclear Fuel Reprocessing and Waste Management (Proc. 4th Int. Conf. London, 1994), Vol. 2, British Nuclear Industry Forum, London (1994).

[192] BUCK, C., SANDERS, M.J., PENGELLY, M.G.A., “Challenges and solutions for the decommissioning of production scale plutonium facilities”, Nuclear Decom ’92 — Decommissioning of Radioactive Facilities (Proc. Conf. London, 1992), Institution of Mechanical Engineers, London (1992) 119–130.

[193] WILLIAMS, J., et al., “Planning and strategy for the decommissioning of nuclear facilities”, ibid., pp. 1–8.

[194] SMITH, R.H., HOOTMAN, H.E., "Dismantlement and decontamination of a plutonium-238 facility at the Savannah River site", Waste Management '94 (Proc. Symp. Tucson, 1984), Arizona Board of Regents, Tucson (1984) 2029–2032.

[195] HINTON, J.P., et al., For Nuclear Explosions, "All Plutonium is Good Plutonium", Proliferation Vulnerability Red Team Report, SAND 97-8203, Sandia Natl Labs, TX (1996).

[196] BURROWS, B., et al., "The safeguarding of MOX fuel facilities in Europe: a reality", Institute of Nuclear Materials Management Annual Meeting (Proc. 37th Mtg Naples, Florida, 1996), INMM, Northbrook, IL (1996) 783–790.

[197] BAER, A.J., "Taming the nuclear dragon: our task for the new century", Peaceful Uses of Nuclear Energy (Proc. Int. Symp. Tokyo, 2000), Osaka University (2000) (CD-ROM).

[198] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Verification of Weapon-Origin Fissile Material in the Russian Federation and the United States, PR 2001/19, IAEA, Vienna (2001).

[199] INTERNATIONAL ATOMIC ENERGY AGENCY, Model Protocol Additional to the Agreement(s) between State(s) and the International Atomic Energy Agency for the Application of Safeguards, INFCIRC/540, IAEA, Vienna (1997).

[200] HEINONEN, O.J., MURAKAMI, K., SHEA, T., "Overview of safeguards aspects related to MOX fuel", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 74–78 (CD-ROM).

[201] ARENAS CARRASCO, J., et al., "Safeguards on MOX assemblies at LWRs", ibid., pp. 418–429.

[202] XERRI, C., et al., "Control of nuclear material hold-up in MOX fuel fabrication plants in Europe", Safeguards and Nuclear Material Management (Proc. 23rd ESARDA Ann. Mtg Brugges, 2001), British Nuclear Energy Society, London (2001) (CD-ROM).

[203] KLOSTERBUER, F.S., KERN, A.E., PAINTER, A.J., TAKAHASHI, S., "Unattended mode operation of specialized NDA systems", Institute of Nuclear Materials Management Annual Meeting (Proc. 30th Mtg Orlando, 1989), INMM, Northbrook, IL (1989) 262–266.

[204] MENLOVE, H., et al., Smart unattended systems for plutonium safeguards, J. Nucl. Mater. **24** 4 (1996) 135.

[205] TOLBA, A., et al., "The unattended NDA measurement systems at MOX fuel fabrication facilities in Belgium", MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 177–180 (CD-ROM).

[206] HASSAN, B., et al., "Continuous monitoring of plutonium solution in a conversion plant", ibid., pp. 181–189.

[207] TAKAHASHI, S., KANEKO, H., HIGUMA, T., OHTANI, T., SEYA, M., "Safeguards system at plutonium fuel facility", Institute of Nuclear Materials Management Annual Meeting (Proc. 31st Mtg Los Angeles, 1990), INMM, Northbrook, IL (1990) 301–305.

[208] TAKAHASHI, S., OHTANI, T., OHSHIMA, H., "Experience in safeguards implementation at the plutonium fuel production facility (PFPF)", International Nuclear Safeguards 1994: Vision for the Future (Proc. Int. Symp. Vienna, 1994), Vol. 2, IAEA, Vienna (1994) 87–100.

[209] ASAKURA, K., OHTANI, T., ABHOLD, E.M., BUCK, E.S., MENLOVE, O.H., “Development of remote monitoring system for the unattended mode NDA in PFPF”, Institute of Nuclear Materials Management Annual Meeting (Proc. 39th Mtg Naples, Florida, 1998), INMM, Northbrook, IL (1998) (CD-ROM).

[210] ASANO, T., TAKAHASHI, S., ABHOLD, E.M., HARKER, C.W., BUCK, E.S., “Development of remote monitoring for unattended mode NDA in PFPF (Phase II)”, Institute of Nuclear Materials Management Annual Meeting (Proc. 40th Mtg Phoenix, 1999), INMM, Northbrook, IL (1999) (CD-ROM).

[211] INTERNATIONAL ATOMIC ENERGY AGENCY, The Structure and Content of Agreements between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons, INFCIRC/153, IAEA, Vienna (1972).

[212] SELLINSCHEGG, D., DENDEL, M., TAKAHASHI, S., Application of Near-Real Time Accountancy (NRTA) at An Automated Mixed Oxide (MOX) Fuel Fabrication Plant, American Nuclear Society, La Grange Park, IL (1991) 98.

[213] YOSHIDA, M., SUZUKI, T., KOBAYASHI, H., OHTANI, T., “Experience of determination of plutonium and uranium contents in MOX fuel by IDMS”, Institute of Nuclear Materials Management Annual Meeting (Proc. 42nd Mtg Indian Wells, 2001), INMM, Northbrook, IL (2001) (CD-ROM).

[214] MENLOVE, O.H., AUGUSTSON, R.H., OHTANI, T., TAKAHASHI, S., ABEDIN-ZADEH, R., “Remote-controlled NDA systems for feed and product storage at an automated MOX facility”, Institute of Nuclear Materials Management Annual Meeting (Proc. 30th Mtg Orlando, 1989), INMM, Northbrook, IL (1989) 267–273.

[215] ASANO, T., KOBAYASHI, H., TAKAHASHI, S., MENLOVE, O.H., WENZ, T., “Development of improved hold-up measurement system at plutonium fuel production facility”, Institute of Nuclear Materials Management Annual Meeting (Proc. 38th Mtg Phoenix, 1997), INMM, Northbrook, IL (1997) (CD-ROM).

[216] MARUYAMA, H., FUJIWARA, S., TAKAHASHI, S., MENLOVE, O.H., WENZ, T., “Advanced material accountancy glove box assay system for process material at PFPF”, Institute of Nuclear Materials Management Annual Meeting (Proc. 42nd Mtg Indian Wells, 2001), INMM, Northbrook, IL (2001) (CD-ROM).

[217] WENZ, T., MENLOVE, O.H., MARUYAMA, H., FUJIWARA, S., TAKAHASHI, S., “In situ Pu isotopic measurements using electromechanically cooled HPGe detectors”, ibid.

[218] GUNNINK, R., MGA: A Gamma-Ray Spectrum Analysis Code for Determining Plutonium Isotopic Abundances, Rep. UCRL-LR-103220, Lawrence Livermore National Laboratory, CA (1990).

[219] INTERNATIONAL ATOMIC ENERGY AGENCY, Guidance and Considerations for the Implementation of INFCIRC/225/Rev.4, The Physical Protection of Nuclear Material and Nuclear Facilities, IAEA-TECDOC-967 (Rev. 1), Vienna (2000).

[220] SCHREIBER, H.J., PIANA, M., MOUSSALLI, H., SAUKKONEN, H., “Integrated software package for nuclear material safeguards in a MOX fuel fabrication facility”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 127–134 (CD-ROM).

[221] SAUVAGE, M., BROMFIELD, A.M., MARTH, W., “Overview on European fast reactor operating experience”, Fast Reactor and Related Fuel Cycles (Proc. Int. Conf. Kyoto, 1991), Atomic Energy Society of Japan, Tokyo (1991) 2.1–1.

[222] ELIE, X., CHAUMONT, J.M., “Operation experience with the Phénix prototype fast reactor”, ibid., 5.1–1.

[223] GREGORY, C.V., “Operating experience with the Prototype Fast Reactor at Dounreay”, ibid., 5.3–1.

[224] FABRE, J.J., ROUCHES, F., “FBR fuel design, manufacture and reprocessing experience in France”, ibid., 15.1–1.

[225] DUCKWITZ, C.A., KRELLMANN, J., MUHLING, G., “German experience with FBR MOX fabrication”, ibid., 15.5–1.

[226] POPLAVSKI, V.M., et al., The use of plutonium in reactor fuel, At. Energ. **89** 4 (2000) 314.

[227] RESHETNIKOV, F.G., BIBILASHVILI, Yu.K., GOLOVNIN, I.S., Development, Fabrication and Exploitation of Fuel Rods for Nuclear Power Plants, 2 Vols, Energoatomizdat, Moscow (1995) (in Russian).

[228] RESHETNIKOV, F.G., BIBILASHVILI, Yu.K., GOLOVNIN, I.S., Investigation of Mixed Oxide Fuel, Journal Issues of Atomic Science and Techniques (VANT), Series: Mater. Sci. New Materials **52** 1 (1995) 6–12.

[229] GOLOVNIN, I.S., Properties of plutonium dioxide as nuclear fuel, At. Energ. **89** 2 (2000) 117.

[230] KATSURAGAWA, M., KASHIHARA, H., AKEBI, M., Status of liquid metal fast breeder reactor fuel development in Japan, J. Nucl. Mater. **204** (1993) 14–22.

[231] BROWN, C., Status of PFR fuel performance and its relevance to future fast reactor designs”, Nucl. Energy **31** 1 (1992) 49.

[232] GOETZMANN, O., “LMFBR-fuel pin behaviour”, Fast Reactor Core and Fuel Structural Behaviour (Proc. Conf. London, 1990), British Nuclear Energy Society, London (1990).

[233] NAGANUMA, M., et al., “High burnup irradiation performance of annular fuel pins irradiated in fast reactor PFR”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 311–321 (CD-ROM).

[234] BROWN, C., et al., “Cladding and wrapper development for fast breeder reactor high performance”, Fast Reactors and Related Fuel Cycles (Proc. Int. Conf. Kyoto, 1991), Vol. 1, Atomic Energy Society of Japan, Tokyo (1991) 7.5.

[235] FORD, I.J., THETFORD, R., “Influence of cladding materials on fuel performance at high burnup”, Fast Reactor Core and Fuel Structural Behaviour (Proc. Conf. London, 1990), British Nuclear Energy Society, London (1990).

[236] PLITZ, H., CRITTENDEN, G.C., LANGUILLE, A., Trans. Am. Nucl. Soc. **68** (1992) 203–205.

[237] BOTTCHER, J.H., INOUE, M., NOMURA, S., “Operational reliability testing of MOX fuels for future FBRs”, Global ’97: Future Nuclear Systems (Proc. Conf. Yokohama, 1997), Vol. 2, American Nuclear Society, La Grange Park, IL (1997) 1086–1092.

[238] STRUWE, D., et al., “Fuel pin destruction modes — experimental results and theoretical interpretation of the CABRI-1 programme”, Fast Reactor Core and Fuel Structure Behaviour (Proc. Mtg Inverness, 1990), British Nuclear Energy Society, London (1990) 127.

[239] SATO, I., IMKE, U., PFRANG, W., PAPIN, J., BERNE, M., “Transient fuel pin behaviour and failure condition in the CABRI-2 in-pile tests” (Proc. Int. Top. Mtg Obninsk, 1994), Vol. 2, IPPE, Obninsk (1994) 2-134–2-145.

[240] JAPANESE GOVERNMENT ATOMIC ENERGY COMMISSION, Long Term Programme for Research, Development and Utilization of Nuclear Energy, Atomic Energy Commission, Tokyo (2000).

[241] ABE, T., et al., “Optimization of fast breeder reactor core design for improving economy”, paper presented at Global 2001 Conf. Paris, 2001.

[242] UKAI, S., et al., Tube manufacturing and characterization of oxide dispersion strengthened ferritic steels, J. Nucl. Mater. **702** (2000) 283–287.

[243] NODA, H., “Current status of fast reactor cycle system in Japan”, ICONE-8 (Proc. Conf. Baltimore, 2000), American Society of Mechanical Engineers, New York (2000) (CD-ROM).

[244] KEMMISH, W., Gas-cooled fast reactors, Nucl. Energy **21** (1982) 77–78.

[245] ABRAM, T.J., et al., “The enhanced gas-cooled reactor”, ICONE-8: International Conference on Nuclear Engineering (Proc. Int. Conf. Baltimore, 2000), ASME, New York, NY (2000) 849–858.

[246] NUKEM, The Pebble Bed Modular Reactor, NUKEM Market Report, RWE NUKEM, Alzenau (2000).

[247] US DEPARTMENT OF ENERGY, Executive Summary of Gas-Turbine Modular Helium Reactor Conceptual Design, USDOE, Washington, DC (1999).

[248] OECD NUCLEAR ENERGY AGENCY, Management of Separated Plutonium — The Technical Options, OECD NEA, Paris (1997) 108–110.

[249] MAYORSHIN, A.A., et al., “Advanced fuel cycle on the basis of pyroelectrochemical process for irradiated fuel reprocessing and vibropacking technology”, MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No. 3/P, IAEA, Vienna (2000) 459–467 (CD/ROM).

[250] OECD NUCLEAR ENERGY AGENCY, Management of Separated Plutonium — The Technical Options, OECD NEA, Paris (1997) 115–116.

[251] CHAWLA, R., et al., “First experimental results from neutronics and in-pile testing of a Pu–Er–Zr oxide inert matrix fuel”, TOPFUEL ’01 (Proc. Int. Conf. Stockholm, 2001), European Nuclear Society, Bern (2001) CD-ROM.

[252] WELHUM, POHL P., Isotopic analysis on PuO_2–ThO_2 fuel irradiated in Lingen BWR power plant, Eur. Appl. Res. Rep., Nucl. Sci. Technol. Sect. **2** 6 (1981) 1552–1577.

[253] GANGULY, C., “Oxide and non-oxide ceramic fuels development and production in India”, Advanced Ceramics-2001 (ISAC-2001) (Proc. Int. Symp. Hyderabad, 2001), Indian Ceramic Society, Central Glas & Ceramic Research Institute, Kolkata (2002) 138–152.

ACRONYMS AND ABBREVIATIONS

ABWR	advanced boiling water reactor
ADS	accelerator driven system
ADU	ammonium di-uranate
AEC	Atomic Energy Commission (USA)
AECL	Atomic Energy of Canada Limited
AGR	advanced gas cooled reactor (UK)
ALARP	as low as reasonably practicable
APM	Atelier pilote de Marcoule (France)
AT	Atelier de traitement (La Hague)
ATGR	high temperature gas cooled reactor
ATR	advanced thermal reactor (Japan), advanced test reactor (Idaho, USA)
ATWS	anticipated transient without scram
AUC	ammonium uranyl carbonate
AUPuC	ammonium uranyl plutonyl carbonate
BARC	Bhabha Atomic Research Center (India)
BNFL	British Nuclear Fuels plc (UK)
BOL	beginning of life
BR	Belgian reactor
BWR	boiling water reactor
C/S	containment and surveillance
CANDU	Canadian deuterium uranium reactor
CEA	Commissariat à l’énergie atomique (France)
CEC	Commission of the European Communities (European Union)
CERN	Conseil Européen de recherche nucléaire (now renamed European Organisation for Nuclear Research) (Geneva)
CFCa	Centre de fabrication de Cadarache (France)
CNA	Centrale nucléaire des Ardennes (France)
CNE	Commission nationale d’évaluation (France)
COCA	Cobroyage (co-milling) Cadarache
Cogema	Compagnie générale des matières nucléaires (France)
COPRECAL	coprecipitation and calcination process for MOX powder production (US)
CPF	Chemical Processing Facility (Japan)
CRL	Chalk River Laboratories (AECL)
DA	destructive analysis
DEMOX	Demonstration MOX fabrication plant (RF)
DF	decontamination factor
DFR	Dounreay Fast Reactor (UK)
DNBR	departure from nucleate boiling ratio

DOE	Department of Energy (USA)
DPA	displacements per atom
DSIN	Direction de la sûreté des installations nucléaires (France)
EDF	Electricité de France
EFPD	equivalent full power days
EOC	end of cycle
ERC	Experimental Research Complex (Dimitrovgrad)
$F_{\Delta H}$	peaking factor on enthalpy
FA	fuel assembly (LWR) or subassembly (FBR)
FBFC	Franco-Belge de fabrication de combustible
FBFC International	Franco-Belge de fabrication de combustible international (Dessel)
FBR	fast neutron reactor (either breeder or burner)
FBTR	Fast Breeder Test Reactor (India)
FCCI	fuel–cladding chemical interaction
FCMI	fuel–cladding mechanical interaction
FEX	Nuclear Fuel Experts S.A. (Belgium)
FFTF	Fast Flux Test Facility
FGR	Fission Gas Release
fima	fissions per metal atom
FR	fuel rod (LWR) or fuel pin (FBR)
GCR	gas cooled reactor
GRANAT	granulation technology (RF)
GTMHR	gas turbine modular helium cooled reactor
HAO	Haute Activité Oxyde (La Hague)
HBWR	Halden BWR (heavy water)
HEU	highly enriched uranium
HFR	High Flux Reactor (Petten)
HLW	high level waste
HM	heavy metal (= U + Pu + Am)
HRGS	high resolution gamma ray spectrometry
HRP	Halden Reactor Project (OECD)
HSE	Health and Safety Executive (UK)
HTR	high temperature reactor
HWR	heavy water reactor
IDMS	isotope dilution mass spectrometry
IFA	instrumented fuel assembly (HBWR)
IIV	interim inventory verification
ILW	intermediate level waste
IMF	inert matrix fuel
INEEL	Idaho National Engineering and Environmental Laboratory (USA)
IPSN	Institut de protection et de sûreté nucléaire (France)

ITU	Institute of Transuranic Elements (Karlsruhe)
JAERI	Japan Atomic Energy Research Institute
JMOX	Japan MOX plant
JNC	Japan Nuclear Cycle Development Institute, previously PNC (Tokai-mura)
JNFL	Japan Nuclear Fuels Limited
Joyo	Experimental FBR (Japan)
KAERI	Korean Atomic Energy Research Institute
KNK	Kompakte Natriumgekühlte Kernreaktoranlage (FBR)
KWO	Kernkraftwerk Obrigheim (PWR)
LANL	Los Alamos National Laboratory (USA)
LEU	low enriched uranium
LLFP	long lived fission products
LOCA	loss of coolant accident
LWR	light water reactor
MA	minor actinide
Magnox	metal fuel in magnesium alloy can (UK)
MDF	MOX Demonstration Facility (BNFL)
MELOX	melange oxides fabrication plant (Marcoule)
MIMAS	micronized master blend (Belgonucléaire)
Monju	prototype FBR (Japan)
MOX	mixed oxide fuel of uranium and plutonium
MZFR	MultiZweck Forschung Reaktor
NDA	non-destructive assay
NDE	non-destructive examination
NEFW	Department of Nuclear Energy/Division of Nuclear Fuel Cycle and Waste Technology (IAEA)
NII	Nuclear Installations Inspectorate (UK)
NIRAS/ONDRAF	Nationale Instelling voor Radioactief Afval en verrijkte Splijtstoffen/ Organisme National des Déchets RAdioactifs et des matières Fissiles enrichies
NITROX	denitration process for PuO_2 production (France)
NOK	NordOstschweizerische Kraftwerke (Switzerland)
NPP	nuclear power plant
NPT	Non-Proliferation Treaty
NRU	National Research Universal reactor (AECL)
NSRR	Nuclear Safety Research Reactor
NSRW	Department of Nuclear Safety/Division of Radiation and Waste Safety (IAEA)
OCOM	optimized co-milling
OECD	Organisation for Economic Co-operation and Development

O/M	oxygen to metal ratio
P&T	partitioning and transmutation
PAKET	Packaging Technology (Russian MOX fabrication facility)
PARALLEX	Parallel Experiment (Canada, Russian Federation, USA)
PBMR	pebble bed modular reactor
PCDF	plutonium finishing line (Tokai)
PCM	plutonium contaminated material
PCMI	pellet–cladding mechanical interaction
PFDF	Plutonium Fuel Development Facility (Tokai-mura)
PFFF	Plutonium Fuel Fabrication Facility (Tokai-mura)
PFPF	Plutonium Fuel Production Facility (Tokai-mura)
PFR	Prototype Fast Reactor (UK)
PHWR	pressurized heavy water reactor
PIE	post-irradiation examination
PIV	physical inventory verification
PNTL	Pacific Nuclear Transport Ltd
POCO	post-operational clean-out
PREFRE	Power Reactor Fuel Reprocessing plant (India)
PSI	Paul Scherrer Institute (Switzerland)
PWR	pressurized water reactor
QA/QC	quality assurance/quality control
R&D	research and development
RETF	Reprocessing Equipment Testing Facility (Japan)
RFFL	Reprocessing Fuel Fabrication Laboratories (AECL)
RIA	reactivity initiated accident
RIAR	Research Institute of Atomic Reactors (also known as NIIAR) (Dimitrovgrad)
RRP	Rokkasho Reprocessing Plant (Japan)
RT	Radiochemical Technology (Russian reprocessing plant)
SAL	Safeguard Analytical Laboratory
SAP	Service d’atelier pilote (Marcoule)
SBR	Short Binderless Route (BNFL)
SCK/CEN	StudieCentrum voor Kernenergie/Centre d’étude de l’énergie nucléaire (Mol)
SGCP	Department of Safeguards/Division of Concepts and Planning (IAEA)
SGOB	Department of Safeguards/Division of Operations B (IAEA)
SMP	Sellafield MOX Plant (BNFL)
SPERT	Special Power Excursion Reactor Test (Idaho)
SQ	significant quantity
SSA	specific surface area
TBP	tri-butyl phosphate

TCM	Technical Committee Meeting (IAEA)
TD	theoretical density
TDN	thermal de-nitration
TEPCO	Tokyo Electric Power Company
THORP	Thermal Oxide Reprocessing Plant (UK)
TIG	tungsten inert gas welding technique
TNB	TransNuBel (Belgium)
TNP	TransNucléaire Paris
TOP	Traitement d'oxydes pilote (Marcoule)
TOR	Traitement d'oxydes rapide (Marcoule)
TRP	Tokai Reprocessing Plant (Japan)
TRU	TRansUranium
UCD	Unité centrale de décontamination (La Hague)
UKAEA	United Kingdom Atomic Energy Authority
UNGG	Uranium naturel graphite gaz (France)
UP2/UP3	Unité de production (reprocessing plants, France)
URP	Unité de récupération du plutonium (La Hague)
WAK	WiederaufarbeitungsAnlage Karlsruhe (Germany)
WPu	weapons grade plutonium
WWER	water cooled, water moderated reactor (Russian type PWR)

CONTRIBUTORS TO DRAFTING AND REVIEW

This Technical Report was jointly prepared by a working group consisting of:

Bairiot, H. (chairman)	FEX, Belgium
Blanpain, P.	Framatome ANP, France
Farrant, D.	British Nuclear Fuels Ltd, United Kingdom
Ohtani, T.	Japan Nuclear Cycle Development Institute, Japan
Onoufriev, V.	International Atomic Energy Agency
Porsch, D.	Framatome ANP, Germany
Stratton, R.	Nordostschweizerische Kraftwerke, Switzerland

Consultants Meetings

Vienna, Austria: 8–9 August 2000, 12–13 December 2000, 17 April 2001,
12–15 June 2001, 16–19 October 2001

The working group also wishes to acknowledge significant inputs from external contributors:

Brown, C.	British Nuclear Fuels Ltd, United Kingdom
Deramaix, P.	Belgonucleaire, Belgium
Golovnin, I.	Bochvar Institute (VNIINM), Russian Federation
Haas, E.	International Atomic Energy Agency
Laraia, M.	International Atomic Energy Agency
Nagai, S.	Japan Nuclear Cycle Development Institute, Japan
Pope, R.	International Atomic Energy Agency
Sanchis, H.	Transnucléaire, France
Shea, T.	International Atomic Energy Agency
Weston, R.	British Nuclear Fuels Ltd, United Kingdom