

# Trends in the Nuclear Fuel Cycle

Economic, Environmental  
and Social Aspects



Nuclear Development

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**Economic, Environmental and Social Aspects**

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## FOREWORD

Nuclear energy has been part of the world's energy mix for almost fifty years. However, over the past twenty years increased public concern over this form of energy has resulted in socio-political constraints on its use. Yet recently expressed concerns over sustainable development and climate change have caused renewed interest in the potential role of nuclear energy in the world's future energy supply mix.

The role of nuclear energy in a sustainable energy future has multiple facets, a significant number of which relate to the nuclear fuel cycle. Many sustainability issues are associated with the fuel cycle, e.g. use of natural resources, economics, waste arisings, public acceptance, proliferation resistance. In addition, the development of new reactor types with improved characteristics will generally entail concurrent important developments in the related fuel cycle.

With this in mind, the NEA Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle considered that a fresh look at the nuclear fuel cycle options was timely. A study was thus undertaken to review the main developments in the nuclear fuel cycle that may further improve the competitiveness and sustainability of nuclear generating systems.

The present report seeks to place the different fuel cycle developments in a medium and longer-term framework that may respond to societal demand for more sustainable energy options. It addresses prioritisation issues by considering criteria and indicators for future nuclear energy systems. It supports other NEA reports on the role of nuclear energy in a sustainable context and is directed towards those readers involved in nuclear energy policy and decision making.

The study was carried out by an Expert Group (see Annex A) comprising representatives of the nuclear industry, government agencies, and research organisations involved in various aspects of nuclear fuel cycle development. It does not necessarily represent the official governmental opinion of NEA Member countries or that of the international organisations involved. The report is published on the responsibility of the Secretary-General of the OECD.

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## EXECUTIVE SUMMARY

Nuclear energy has been part of the world's energy mix for almost fifty years. However, over the past twenty years increased public concern towards this form of energy has resulted in socio-political constraints on its use. While over these past twenty years the world was able to cope with an increasing energy demand by relying more on fossil fuels, recently concerns over climate change have caused renewed interest in the potential role of nuclear energy in the world's future energy supply mix.

Nuclear energy has special characteristics with respect to sustainable development, which differ significantly from other sources of energy production. A significant part of its specific characteristics relate to the nuclear fuel cycle. This report aims towards giving a description of the developments and trends in the nuclear fuel cycle that may improve the competitiveness and sustainability of nuclear generating systems. It is intended to support other NEA reports on the role of nuclear energy in a sustainable context, that were published as input to the OECD horizontal project on sustainable development. As such, it is directed towards those readers involved in policy and decision making on nuclear energy development and deployment.

An Expert Group comprising representatives of the nuclear industry, government agencies, and research organisations involved in various aspects of nuclear fuel cycle development has prepared the report. The Group recognised from the start that any description of developments and trends in the nuclear fuel cycle also needs to give special attention to non-technical aspects, in particular to sustainability dimensions.

After a brief introduction in Chapter 1, presenting the concept of sustainable development and setting nuclear energy and the nuclear fuel cycle in this perspective, the report characterises, in Chapter 2, the world's resources of nuclear fuel materials and the demands made on those resources by the nuclear power programmes. The yearly uranium demand to sustain the present nuclear power capacity, i.e. some 430 reactors world-wide, is about 60 000 tonnes uranium. The "Known Conventional Resources" support such a demand for some 80 years. The "Undiscovered Conventional Resources" could extend that operating period by a factor of three, while the use of recycling and fast breeder reactors could alleviate the use of these resources a hundred-fold. In parallel, non-conventional resources and new technologies, such as the introduction of thorium fuel cycles have the same potential. Ultimately, uranium extraction from seawater (with the potential of some 4 000 million tonnes) would constitute a virtually unlimited source of supply provided that its development, limited today to a laboratory scale, would be needed and become economically viable.

The report further describes the various constitutive steps of today's nuclear fuel cycle in Chapter 2 and developments in the nuclear fuel cycles in light of improving its competitiveness and sustainability in Chapter 4. Where some of these developments are part of a longer-term endeavour, e.g. development of a fully recycling system, many ongoing industrial development programmes in different areas of the nuclear fuel cycle comprise important environmental improvements in addition to economic enhancements.

The extraction and processing of uranium, for example, has environmental impacts through mill tailings and the release of radon. Specific industrial mining programmes therefore were developed to improve mining techniques and environmental measures and to reduce these impacts to very low levels, comparable to the natural background radioactivity.

In the field of uranium enrichment, the development of centrifuge technology has led to a reduction of costs, mainly due to a reduction of the energy consumption by a factor of 50 compared to gas diffusion technology. This process likely will dominate the enrichment field in the medium term. However, laser enrichment should not be ruled out in the longer term, as it allows for selective re-enrichment of reprocessed uranium.

There is continuous improvement towards higher performance, reliability and safety of fuel design and fabrication. Directions in fuel optimisation address the integration of front-end and back-end issues together with improving the operational performance of the nuclear power plants. In that respect, innovative fuel forms are in development, partly in the framework of a life-cycle approach, to bring reductions in spent fuel quantities and hence in long-term liabilities, better resource conservation and at the same time improving plant availability and fuel cycle costs. While the relative economic merit of reprocessing may vary over time, it holds the potential to reduce the specific consumption of uranium as a natural resource and especially to reduce the quantities of radio-nuclides to be disposed of.

In the back-end, the waste disposal solutions proposed today are mainly country specific with every country developing its disposal facilities adapted to its own geological substrate, because of the principle that the producer of waste has to take care of its wastes. Several such facilities have been proposed for the ultimate disposal of high-level long-lived waste, reaching a stage of advancement where the scientific and technical experts feel confident of their feasibility and safety. However, limited societal and political consensus has postponed the implementation of these solutions.

Although the Group cannot claim to be fully representative of all the many interested parties involved in decision making on nuclear energy and other energy options, the Group attempted, by taking an approach broader than only the economic and technical aspects, to lay a basis for a more comprehensive decision-aiding technique, i.e. multi-criteria analysis (MCA), that can be used in such decision making. Representatives of different parts of society (stakeholders) could use this technique to assess the sustainability characteristics of a particular project. MCA is based on the identification of the criteria associated with a sustainable development approach and of their key quantifying indicators. This methodology and the selected criteria and indicators for the assessment of nuclear fuel cycle developments are presented in Chapter 3.

While the report does not intend to provide a unique merit scale classification of the potential fuel cycles in use or in development as this is the province of policy makers and other stakeholders, Chapter 5 gives an overview of potential and promising nuclear systems involving reactors and their associated fuel cycles in the medium to long-term future in relation to further reactor developments which may respond to public concerns. The main advantages and also the perceived drawbacks are identified. Whilst the HTGR fuel cycle is essentially a revival of a well advanced former concept which may reach industrial maturity within the next ten to twenty years, other advanced fuel cycle developments rely on new reactor concepts which may well further reduce the long-term environmental consequences of nuclear power. These new concepts will require, however, substantial long-term R&D efforts in the fuel cycle and will likely take decades to implement; e.g. thorium-fuel cycle, molten salt reactors and partitioning and transmutation of long-lived radio-nuclides. Further, some of these fuel cycles, essentially those based on partitioning and transmutation, may also need operation for decades or even centuries to reach equilibrium and to really achieve the claimed significant reduction of the potential radio-toxicity of the waste.

The Expert Group, after compiling and reviewing the information and data described in the report, draws the following conclusions (see Chapter 6):

- When viewed from a large set of criteria, nuclear power shows a unique potential as a large scale sustainable energy source. Some new reactor types, like the HTGRs and their associated fuel cycles will open new nuclear applications beyond today's electricity production, e.g. production of hydrogen, which may significantly extend the role that nuclear energy may play in a sustainable development future.
- A global market exists for the front end of the nuclear fuel cycle but the back-end remain essentially developed on a national level. The front-end facilities will likely be adequate in the next decade, but a problem of the adequacy of facilities beyond this decade, especially if nuclear power is to pick up momentum, is recognised.
- Governments and industry have developed appropriate environmental protection measures in the nuclear fuel cycle, including transport. No major technical problems remain and current fuel cycles may essentially be seen as a mature business activity with a very low impact on the environment.
- The nuclear fuel cycle still holds a potential for further economic and environmental optimisation. Major developments in several fields, e.g. fuel performance (burn-up), reprocessing techniques, waste disposal concepts, are being investigated.
- Disposal facilities in varied geological substrates (salt, clay, granite and tuff) have been proposed for the ultimate waste disposal of high-level long-lived waste, reaching a stage of advancement where the scientific and technical experts feel confident of their feasibility and safety. However, societal consensus about the final implementation of these solutions has not yet been reached.
- A major issue arises in relation to the future funding of long-term R&D in order to deploy the full potential of nuclear energy as part of the future energy mix. The reduction of funding of nuclear R&D by governments as well as the limited potential by industry in today's competition oriented market context to fund long-term R&D, asks for a new reflection on the role of governmentally funded R&D. In essence, increased international collaboration of government-funded nuclear R&D is needed in order to fully develop advanced nuclear options within available funding constraints.
- The gap between the public perception and the expert's opinions calls for improved information and public (stakeholder) participation. The multiplicity of aspects involved in sustainable development calls for the establishment of a common set of indicators of such development. The multi-criteria analysis of different fuel cycle options may be an appropriate tool to improve public confidence by letting people participate in the assessment of options. There is however a need to improve the quantification of the criteria and indicators and the availability of data for new advanced fuel cycle developments before this methodology may actually fully apply.



## 1. INTRODUCTION

The availability of energy in large quantities and at acceptable cost is a crucial element of development for modern societies, but energy production and use involve trade-offs among economic, social and environmental factors. Demographic pressure and industrialisation in the developing countries are leading to significant new energy demands, met today mainly by fossil fuels which release greenhouse gases in the atmosphere. Environmental, economic and health concerns related to burning fossil fuels will contribute to an increased demand for strong energy conservation measures and, ultimately, for new carbon-free energy technologies. Integrating these conflicting economic, social and environmental factors in a globally balanced trade-off approach makes up the concept of sustainable development.

Today, governments focus mainly on demand control, hydroelectricity and renewable energy sources to limit pollution and greenhouse gas releases. However, the potentials for demand control and hydroelectricity are limited, and renewables are diffuse and currently are expensive. To many experts, nuclear energy appears to be the only fully proven technology that would be able to supply large amounts of electricity without the release of greenhouse gases, sulphur or nitrous oxides, that cause acid rains and urban smog and are of concern for climate change. Indeed, nuclear energy already is avoiding some 8% of greenhouse gases, that would be emitted if the current nuclear electricity generation would be replaced by fossil fuel fired generation. However, its use is questioned by a part of the public and some government officials because of concerns about availability of fuel reserves, costs, reactor-accident risks, radioactive-waste management and disposal issues, and potential links to nuclear weapons proliferation. The nuclear industry is keenly aware that regaining public and government support is a necessary precedent for a “renaissance” of nuclear power. This implies that all aspects (economic, social, environmental) will have to be assessed in a broad, comparative framework, presented clearly and transparently to the public, and openly debated in order that a new consensus can be sought, based on the sensitivities of different stakeholders.

While the global issue of nuclear energy with respect to other forms of energy production is developed more generically in other OECD/NEA reports [1,2], the report presented herein focuses on the nuclear fuel cycle issues. Indeed, many sustainability issues are associated with the fuel cycle – e.g. pressure on natural resources, economy, waste releases to the environment, public acceptance, proliferation resistance, to name only a few. In addition, the developments of new reactor types with improved characteristics in some of those aspects generally will entail concurrent important developments in the related fuel cycle.

### 1.1 Sustainable development

A key challenge of sustainable development policies is to address its three dimensions (economic, social, environmental) in a balanced way, taking advantage of their interactions and making relevant trade-offs where needed. Energy has links with all three dimensions of sustainable development, as energy services are essential for economic and social development and energy production and use can lead to environmental effects. The role of nuclear energy in a sustainable development context has been reviewed by NEA [1], and it was concluded that nuclear energy has a number of specific

characteristics that respond favourably to the key concept of sustainable development, namely, “*a development that meets the needs of the present without compromising the ability of future generations to meet their own needs*”.

A central goal of sustainability is to maintain or increase the overall assets (natural, man-made and human or social) available to future generations. This means that, in addition to optimising the use of depletable natural resources for energy production, human and man-made resources should be used in the most effective and profitable way. The amount of energy per unit of natural resource consumed may be one important indicator, while the amount of manpower and quality of work may be another which also pertains to different sustainability dimensions. The Expert Group attached importance to the development of nuclear energy as a means to broaden the natural resource base available for energy production to satisfy the needs of current and future generations.

Technological development is essential in this context as it is critical for the support of economic development. However, careful control and monitoring should be in place in order to make technological development consistent with the goals of sustainable development. Technology has made nuclear energy competitive, by current standards, in today’s deregulated electricity markets, since the forward costs of production are low compared with fossil-fuel alternatives. This competitive position is robust from a sustainable development perspective since most health and environmental costs of nuclear energy already are internalised in the production costs and passed on to consumers through the electricity tariffs. Future developments of nuclear energy, being a technology intensive discipline, should therefore work to strengthen this position and, in addition, to address public concerns and to meet the social objectives of sustainable development.

Today, public concerns about the potential impacts of energy production are assuming growing importance. Therefore, it is essential to include the public in a decision-making process through which it can gain confidence that its concerns are being heard and addressed in the current and proposed technological developments. Sustainable development policies in the energy sector will rely on comparative assessment of alternative options taking into account their economics, and their environmental and social impacts, at local, regional and global levels. It is therefore of paramount importance that the nuclear community be fully involved in this exercise and that it identifies and undertakes those developments that have potential to become acceptable to the public, while achieving the technical and economic performance targets.

The search for common indicators to compare all aspects of different means for energy production (e.g. nuclear versus coal, one nuclear system against another) is complicated by the variety of economic, social and environmental impacts to be considered. Impacts may be different in geographical extent (e.g. local, regional, global), in duration (e.g. short-term, long-term, inter-generation), and in phenomenology (e.g. health, environment), and thus their measurement and valuation may be very difficult. Nonetheless, individuals, firms and governments must make decisions, and this means that they have to carry out some kind of valuation, no matter whether it is explicit, implicit or based simply on personal preferences. The goal of explicit valuation is to make the factors going into decision making more transparent and more valid for use in quantitative comparative assessment studies. Indeed, the currently existing reactors and fuel cycle operations, presented in Chapter 2, were developed in response to specific needs, and in the context of the quantitative evaluation criteria that applied at the time they were being developed. Today, however, there is a need to apply different criteria, in particular taking into account the high importance currently given to the issues discussed above. In this connection, Chapter 3 presents and discusses evaluation criteria and assessment techniques that were deemed by the Expert Group to be most relevant for assessing nuclear power and its fuel cycle in a sustainable development context. Chapter 4 then describes various options for nuclear reactor and fuel cycle technologies, while Chapter 5 presents a number of complete fuel cycle schemes, comprising different combinations of the reactor

and fuel cycle technologies covered in Chapter 4. In both chapters, technical and economic data are presented that potentially can be used to quantify these technologies and fuel cycle schemes in terms of the criteria and indicators discussed in Chapter 3.

## 1.2 The case of nuclear energy

After buoyant beginnings, the nuclear industry is today at a crossroad. On one side, it has achieved a significant share of the electricity production (17%) and energy production (7%) in the world. At the end of year 2000, there were more than 430 nuclear reactors producing electricity in the world, with the large majority (92%) being light water reactors, a few heavy water reactors (4%) and some gas cooled reactors (4%). They all use uranium as fissile material, though some also use recycled plutonium as part of their fuel. On the other side, the industry faces strong public concerns about risks of accidents, radioactive waste management and proliferation, in spite of the fact that the industry is subject to a rigorous system of strict controls and prior licensing approvals by national regulatory authorities and international bodies. This regulatory system has resulted in the safe operation of commercial nuclear power reactors, with only one reactor accident with large radiological consequences to the public since the beginning of their deployment (i.e. Chernobyl, a non-LWR reactor type).

Despite an anticipated doubling of the current world electricity generating capacity between today and 2020, only a small number of new reactors currently are under construction. Deregulation of the electricity sector has had the effect of pooling reserve generating capacities on a larger scale than before, thereby postponing the need for new generating capacity of any type and thus decreasing the demands for large-capacity nuclear plants. In the present deregulated environment in the United States and in Europe, nuclear power has to compete on (mainly short-term) economic grounds. For existing nuclear plants, the low fuel cycle costs are helping to keep nuclear generation highly competitive with other options.

Decisions on R&D funding and new plant investments are taken by managers based on the expected return on investment, subject to the conditions and rules of the time. The high capital costs of nuclear plants place a heavier burden on the financing capacities of the generating companies than does the lower capital cost of gas-fired generation. This could slow the growth potential of nuclear power in a competitive deregulated market, in particular if the full costs to society, including issues of sustainability, are not “internalised” in the generation costs attributed to different options.

Yet, although many people had been predicting closure of a significant number of operating nuclear plants and a rapid fall in nuclear electricity production, the situation has reversed dramatically. Indeed, existing nuclear reactors are very competitive on a forward cost basis,<sup>1</sup> and there now is a fierce competition for purchasing the few existing nuclear plants that are offered for sale by some utilities, mostly in the USA. Lastly, many nuclear plant operators are seeking to renew or extend their operating licences in light of the low production costs. It can therefore be expected that the future of existing nuclear plants is reasonably secure and that most of them will continue to be operated as long as it will be feasible on technical and regulatory grounds, with plant lifetimes extended well into the twenty-first century.

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1. US nuclear plant forward production costs averaged 1.83 cent per kilowatt-hour in 1999, lower than coal (2.07 cent), oil (3.18 cent) or gas (3.52 cent), *Nucleonics Week* January 11, 2001 p 3.

Economic efficiency however is only one of the components of sustainable development. Other key issues are resource utilisation efficiency, effective management of potential environmental hazards (emissions and waste), human capital and social dimensions. These dimensions are essentially the domain of governments and international institutional agreements.

Most of the environmental impacts from fossil fuels, notably the potential for climate change from greenhouse gas emissions but also much of the regional air pollution, come from routine operations, and are “externalities”, that is, their consequences are borne by the “common”, not by the “polluter”. By contrast, nuclear energy has a low impact from routine operations, but carries the potential risk, however small, of a severe accident leading to off-site radiation releases.

Radioactive emissions and liquid discharges from nuclear power plants and fuel cycle facilities are regulated strictly and kept below levels at which health or environmental risks may arise. Indeed, the regulatory objective for the nuclear fuel cycle is to confine the bulk of the waste products, allowing the release only of such quantities (subject to rigorous controls) as are considered to have a negligible impact on health and the environment. At present, the dose commitment from the entire nuclear power industry is more than 300 times lower than that from the natural background radiation, and there is a trend to decreasing radioactive emissions per kWh [3], owing to continuing improvements in operating procedures, plant design changes and fuel cycle developments.

Significant releases would occur only under severe accident conditions, which are made extremely unlikely through carefully controlled design and operation of the facilities as well as increased implementation of inherent safety features.

In the time that has passed since the Kyoto protocol in 1997, very little has been achieved to effectively curb CO<sub>2</sub> emissions. The public appears to support the call made at Kyoto for a sizeable (some 12% globally) reduction from today’s emission levels by 2008-2012, that is, only ten years from now, but at the same time is not ready to modify its habits in order to achieve this target. Furthermore, world economic competition “globalisation” limits the tools that an individual government may use to discourage CO<sub>2</sub>-emitting activities, such as fossil fuel burning. For example, carbon taxation may be economically risky when not all countries agree to similar measures, and in any case the resultant increase in costs of fossil-based energy (e.g. coal, oil and natural gas for electricity production) would be unpopular with the public, even though helping to meet CO<sub>2</sub> reduction targets supported by the public. Also, the pace of making drastic changes to the mix of technology types used for energy production, owing to the long lifetime and slow turnover of plant capacity, is such that some 30-40 years would be needed to replace fossil generation by non-polluting energy production, even assuming that new carbon-free alternatives would be developed in time and would be economically competitive.

Therefore, support for a “renaissance” of nuclear power may come when realisation spreads that nuclear energy is truly safe and clean and that there is no alternative to nuclear as a carbon-free energy source that can be deployed in the first half of the 21<sup>st</sup> century on the scales necessary for limiting greenhouse gas emissions, while providing energy at acceptable costs.

Currently, in OECD countries with nuclear power programmes, spent nuclear fuel and nuclear fuel wastes produced under licence are in safe storage, either in pools or in dry storage canisters. The volumes are small, and such interim storage is safe, cheap and reliable, and can be continued on for many decades.

Geological disposal is in various stages of development in different countries, and can be shown to have the potential to provide the required levels of long-term isolation of the wastes to ensure no

effects on human health or the environment. Moreover, this disposal can be made reversible, if the population would so require. The disposal costs are provisioned by the utilities, under supervision by their respective authorities, according to the “Polluter Pays Principle”, and are accounted for by a fraction of the cost of nuclear electricity, which varies from country to country and which is borne by the electricity consumer.

Other developments in each part of the nuclear fuel cycle are ongoing and aim at a reduction of costs, improvement of environmental aspects and/or optimisation of technical aspects in order to improve operational conditions and safety levels. Most of these developments are rather short-term and funded by industry while more long-term developments, funded mainly by governments, aim at improving the nuclear fuel cycle in a more drastic or integral way, e.g. R&D on fast breeder reactors, partitioning and transmutation systems.

It should be remarked, however, that this longer-term R&D has experienced a strong reduction in governmental funding and that lack of adequate funding may jeopardise the long-term options for nuclear energy. Taking into account the high level of funding and long time needed to bring a concept to commercial deployment, there is a need to focus R&D efforts on those concepts that respond best to the identified most critical needs for improvement of the nuclear fuel cycle. However, achieving a consensus on this focus will be difficult as long as there is a divergence among countries and research teams on the criteria adopted for assessment of the potential of different concepts. Therefore, steps need to be taken to reach agreement on the most appropriate set of criteria and assessment tools, as discussed in Chapter 3 of this report.

### **1.3 The nuclear fuel cycle in perspective**

The nuclear fuel cycles in use today are the result of four decades of technological development aiming at the establishment of a reliable, secure, safe and cost-effective energy source. However, the basic elements of these fuel cycles were established early in this period, when the “ground rules” and development objectives were different from those existing today. Therefore, it is deemed timely to take a fresh look at developments and trends in nuclear fuel cycles which may enhance the sustainability of nuclear energy.

In the seventies, nuclear power was perceived as the solution to dwindling fossil fuel reserves, but uranium also was then considered as a scarce commodity. Many decisions made at that time still affect the fuel cycle industry today. To meet the needs of an already large military programme, and in face of the anticipated rapid growth of nuclear production, large fuel cycle facilities were constructed in the mining, conversion and enrichment stages of the fuel cycle, and reprocessing facilities were constructed to provide plutonium for fuelling the expected introduction of breeder reactors. Although breeder reactors have not been deployed commercially, owing to their higher costs relative to LWRs, they continue to be viewed with interest because of their potential to increase the size of the economically useable uranium reserves a hundred-fold.

The slowdown in civilian nuclear power programmes that has occurred since the eighties, together with the agreements reached for reducing nuclear weapons programmes, has led to the current situation where the production capacities of fuel cycle facilities, with the exception of uranium mining, exceed demand.<sup>2</sup> Current demand for natural uranium amounts to 60 000 tonnes per year. Stockpiles

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2. Uranium production amounts now only to 60% of requirements, but since the end of the cold war in the late eighties large amounts of uranium have been freed from military stockpiles, and are available to make up for any imbalance between production and demand over the next ten to fifteen years.

and known uranium resources represent some 70 years of consumption by the present reactors, and the actual uranium resources are thought to be much larger (conventional uranium resources are estimated to be about 15 million tonnes, representing some 250 years of present consumption). With such reserve levels, there is little economic incentive to explore for uranium today, when there may not be a market for it in the next several decades. Unconventional resources are estimated to be about 22 million tonnes, and the world's oceans contain some 4 000 million tonnes (60 000 years of consumption). Significant efforts were deployed in the past to develop a process of extracting uranium from seawater, which was deemed feasible at a cost between 300 to 700 USD/kgU, roughly ten to twenty-five times today's price [4]. As natural uranium represents only about 5-10% of the forward cost of nuclear generated electricity, extraction of uranium from the oceans would provide a virtually unlimited supply, while only doubling the cost of electricity production by the current reactor types, and with a much lower cost penalty if the uranium would be used to fuel advanced reactors, such as the breeder reactor. This vast resource potential, the fast breeders and the thorium cycles still are viewed today, as they were at the birth of the "nuclear era", as the three major routes to secure virtually unlimited resources for nuclear power, thus enabling it to meet the first criteria for sustainable development, i.e. sustainability from the point of view of the earth's resource base. All three routes will require R&D to develop the fuel cycle and reactor technologies necessary for their implementation.

The scientific and technical community generally feels confident that there already exist technical solutions to the spent fuel and nuclear waste conditioning and disposal question. This is a consequence of the many years of work by numerous professionals in institutions around the world. There has been a free exchange of information and knowledge between these professionals and a strong tradition of open documentation available for peer and public review. Within the technical community of waste management experts, the number of sceptics is relatively small whereas there is a wide consensus on the safety and benefits of geologic disposal. Yet, the scientific and technical community faces the resistance of the public. Safe and cost-effective long-term waste management must not only be demonstrated from the technical perspective, but also there needs to be acceptance by the public.

Two waste management options most often considered in public and scientific debates today are extended storage of the waste (rather than irretrievable disposal) and partitioning and transmutation of long-lived radio-nuclides contained within the waste. Partitioning and transmutation involves processing the waste to extract the long-lived radio-nuclides, which then are irradiated in a nuclear reactor or an accelerator driven system to yield products with shorter half-lives, thereby reducing the time required for their isolation from the environment. Special industrial facilities would have to be built and operated over long time periods in order to achieve this result. In any case, it is recognised that it would not be feasible to apply this technique to all types of waste, so that some quantities of radioactive materials still would require long-term isolation. Although both options might be components of an overall waste management strategy, and extended storage over a few decades is already planned in some countries, neither option completely avoids the need for some final disposal route, such as a geologic repository, to be implemented eventually.

Therefore, scientists and managers responsible for developing solutions to the waste management problem generally remain convinced that progress should continue toward implementation of permanent disposal. The main reason is still that of resolving the issue and not leaving it to future generations. Another reason is that the wastes are more exposed to risks of various kinds while on the surface and it would be better to put them underground sooner rather than later.

During the past ten years, significant progress has been made in scientific knowledge and in developing the technologies required for geologic disposal. This includes: a better scientific understanding of the processes which determine the effectiveness of repositories in isolating the waste over long periods; improved characterisation and quantitative evaluation of the ways in which the

engineered barriers and surrounding rock contribute to safety; specific investigations at candidate sites; and experience with practical aspects of underground repository engineering and implementation.

It is, however, generally felt that a fresh look at the nuclear fuel cycle options may be worthwhile in order to take full account of the possible interactions between the different steps and developments in the fuel cycle. Figures of merit or criteria and indicators for future nuclear energy systems are, in that respect, very important, especially as future development of advanced or new nuclear fuel cycles will demand increasing financial resources and government funding may not be increased proportionally.

This report therefore tries to place the different developments in the framework of medium-term and longer-term fuel cycle schemes that may respond to societal demand for more sustainable energy options.

## 2. THE FUEL CYCLE CHAIN

### 2.1 Introduction

Today's nuclear fuel cycle is the result of half a century of development aimed at establishing an economic, environmentally friendly and safe energy source. During that time, the conditions influencing development of the fuel cycle have changed, as has the relative emphasis on different fuel cycles.

Developments in the fuel cycle were driven initially by non-civilian needs for nuclear materials, and later by the choice of nuclear power plant type for civilian applications. Although a number of different reactor concepts were introduced in the early 1950s, the LWR-line emerged as the main line within most OECD countries, plus the PHWR-line in Canada. Other reactor types, such as gas-cooled and fast reactors, were constructed but found limited commercial application. Depending mostly on political choices, the fuel cycle options that were developed differed in essence according to the treatment given to the spent fuel: i.e. the open (or once-through) fuel cycle (OFC) in which the spent fuel is treated as waste, and the closed (or recycling) fuel cycle (RFC) in which the spent fuel is reprocessed and the recovered plutonium (and possibly the uranium also) is recycled. Both fuel cycle options were developed and both have become mature technologies.

A slow-down of nuclear programmes became obvious in the 1980s due to reduced economic growth expectations in the OECD countries, compounded by the reluctant public attitude after the Three Mile Island and Chernobyl accidents. This change of attitude also resulted in a decrease of interest by the younger population for nuclear education and an overall decrease in governmental nuclear R&D budgets.

The future of the nuclear energy option has to be envisaged from a triple perspective:

- Socio-political constraints and environmental concerns.
- Economic market pressure driving nuclear business reforms.
- Nuclear education and infrastructure.

#### *Socio-political constraints and environmental concerns*

In the OECD Member countries, socio-political changes, mainly growing environmental concerns, have been a major trend affecting the scene for the nuclear energy option. The major areas of discussion in almost all Member countries are related to the overall role of nuclear energy in the energy mix, along with considerations on the back-end of the fuel cycle (open or closed). While some Member countries (especially in Europe) changed their policy towards a gradual phase-out of nuclear,

others introduced additional (regulatory and economic) constraints on nuclear energy. The OSPAR<sup>3</sup> convention in Europe introduced the need to strive towards “near-zero” impact; i.e. reducing the radiological effects of any activity down to a level where one could expect to have zero impact on humans and the environment [5].

In spite of these socio-political constraints, the nuclear option is increasingly benefiting from a new awareness of the role that different energy sources play with respect to sustainable development and, in particular, in meeting the agreed reductions of greenhouse gas emissions attendant to the Kyoto protocol. There is increasing concern among the public that these environmental issues, until recently considered as minor and long-term, are important and have to be handled in the near-term. Several independent reviews by international organisations [6,7], national scientific academies [8] or expert groups [9,10], as well as public opinion polls, have concluded that nuclear power will be a necessity for any sustainable development of energy production in the long-term and that, today, nuclear is the only technologically proven large-scale energy source that could supply environmentally friendly energy to a growing world population.

### *Economic market pressure and nuclear business reforms*

Within a deregulated market, utilities are compelled to decrease their electricity generating costs in response to increasing competition. For existing plants, the operation and maintenance (O&M) costs as well as the fuel cycle costs are essentially the only cost elements on which utilities can achieve reductions. The other costs, i.e. capital costs or taxes,<sup>4</sup> are mainly sunk costs or not under the control of utilities, although some gains have been made through plant power uprating. In general, the O&M costs have been stabilising or decreasing during recent years, but further reductions are constrained by safety and regulatory considerations. In addition, O&M costs tend to increase with plant age, a trend which is putting extra emphasis on the fuel cycle cost component.

In the face of a low growth market, consolidation of activities by horizontal and/or vertical integration of companies is a trend which started by the mid-90s and is still ongoing.<sup>5</sup> These mergers aim to improve the economic viability of the consolidated entities by increasing their market share and economies of scale, or by finding synergy in complementary activities.

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3. The Convention for the Protection of the Marine Environment of the North-East Atlantic was opened for signature at the Ministerial Meeting of the Oslo and Paris Commissions, Paris, 21-22 September 1992. The Convention has been signed by all Contracting Parties to the Oslo Convention and to the Paris Convention. See also <http://www.ospar.org/>.
  4. Recent evaluations indicate that about a 15~25% reduction in capital costs of new nuclear power plants could be achieved through evolutionary changes in plant size, construction methods and schedule, design improvements and standardisation (series production), procurement and contractual aspects, and regulation and policy measures [11]. Such reductions in capital cost compare to generation cost reductions for gas-fired power plants of 16 to 54%, and for coal-fired power plants of 3 to 34%, during the past 7 years [12].
  5. In the fuel fabrication area, a merger was announced in early 1999 between General Electric, Toshiba and Hitachi. Recently, the same happened for an even broader field of activities by merging of the nuclear activities of Framatome and Siemens in a new company called Framatome-ANP. Another example is the acquisition of Westinghouse by BNFL and the merging of the former with ABB Nuclear. Late in 2000, the French nuclear industry was reformed by regrouping nuclear activities (COGEMA, Framatome, CEA-Industrie) with other non-nuclear activities under a holding structure called ArevA.

## *Nuclear education and infrastructure*

Finally, technological development on its own enables continuous improvement to the NPPs and to the associated fuel cycle operations. Today's short-term economics, together with the already mentioned socio-political conditions, do not favour new major long-term developments. A remarkable decrease in overall nuclear R&D funding has been in place since the end of the 1980s, while only few initiatives (e.g. in the USA, Japan, France) exist to counter this trend.

This evolution affects the conservation of nuclear skills and infrastructure. A recent report by NEA [13] indicated that a reduced interest for nuclear education could result in problems, as the capability for nuclear development could be weakened in the middle to long-term. The same conclusion holds for nuclear industrial and R&D infrastructure. In the light of these considerations, technological development of major new options for the nuclear fuel cycle could be hampered. International collaboration and financing to pool the existing technical and/or economic resources would in that respect be highly recommended.

This chapter focuses on a brief review of today's status of the nuclear fuel cycle, highlighting the achievements but also the remaining questions in the light of the changes in the energy scene, as described above. It highlights some of the current economically driven industrial trends towards improving performance of the nuclear fuel cycle.

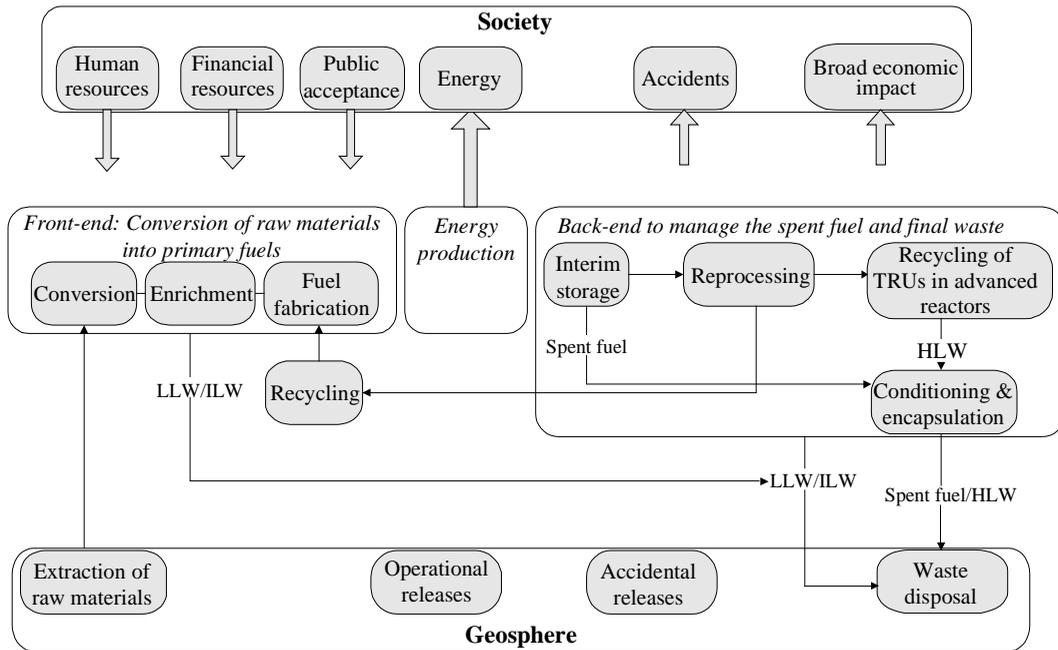
Chapter 4 complements this chapter with a description of the more extensive, longer term and multi-faceted trends in developing new technological options, or new fuel cycle strategies, in order to comply with the perceived increasing need for nuclear power in the 21<sup>st</sup> century and beyond, taking into consideration criteria such as those discussed in Chapter 3. Although this report does not rank the different developments against each other (i.e. it does not attempt to pick a "best" fuel cycle), it does attempt to show how the nuclear community is responding to the demands from society in order to deliver a safe, economic and environmentally friendly source of energy.

## **2.2 The nuclear fuel cycle**

The nuclear fuel cycle (see Figure 2.1) can be viewed as a chain of value-adding steps, each having very particular characteristics and each interacting with the geosphere and with society. Interaction of the fuel cycle with the geosphere relates essentially to the extraction of raw material and the final storage of waste, as well as to operational and accidental releases of radioactive and non-radioactive residuals. The interaction with society consists in essence of delivering energy to the society by use of financial and human resources, as well as the possibility, but low probability, for broad economic impacts to society in the event of a serious accident.

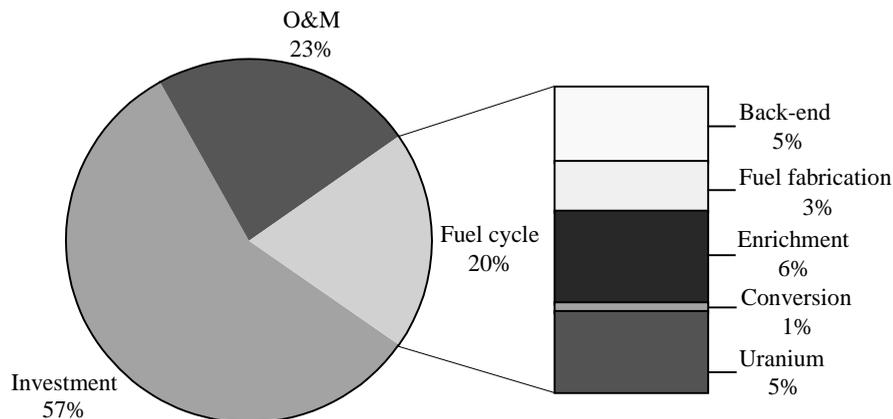
Each fuel cycle step has been economically optimised in itself, as today's competitive pressure exerts an ever-increasing demand on the fuel cycle companies to continuously improve their processes and business. The increasingly stringent regulation guarantees that all fuel cycle steps comply with a constantly decreasing environmental impact.

Figure 2.1 Systems approach to the nuclear fuel cycle



Although a breakdown of the costs of the steps in the nuclear fuel cycle is country specific, Figure 2.2 gives a typical distribution of these costs and shows that the fuel cycle cost accounts for about 20% of the total nuclear electricity generation cost. As a share of the total electricity generation cost, the extraction of uranium ore accounts for only about 5%, the transformation of the raw materials to primary fuels for about 10%, and spent fuel and waste management for about 5% [12, 14].

Figure 2.2 Nuclear electricity generation cost breakdown<sup>6</sup>



6. This breakdown is only indicative; e.g. a low uranium cost (for instance  $\leq$ USD 40/kgU) will result in an even lower fraction in the electricity generation cost.

## 2.2.1 Extraction of raw materials from the environment

Every nuclear fuel cycle starts with the extraction of the fissionable raw materials (uranium or thorium) from the environment, that is, from the earth's crust. Industrial experience until now is mainly with the extraction of uranium to supply the uranium fuel cycle, whereas thorium has essentially remained an untapped potential resource.

### 2.2.1.1 Uranium

#### *The size of the uranium resources*

An essential element in assessing the sustainability of nuclear energy is to understand the size of the resources and the demands that nuclear energy places on them. An extensive analysis of this question is performed regularly by a joint OECD/NEA-IAEA expert group, which publishes a report on a biennial basis, known as the Red Book [15], the results of which are summarised in this section.

Uranium is widely distributed in the earth's crust. Conventional uranium resources are estimated at around 4 million tU (Known Conventional Resources) recoverable at costs  $\leq$ USD 130/kgU. The amount of Known Conventional Resources includes about 3 million tU of the category Reasonably Assured Resources (RAR), recoverable at costs  $\leq$ USD 130/kgU [15]. The conventional resources alone could sustain today's nuclear power capacity for about the next 60 years. In principle, these would last over 60 times longer if uranium and plutonium present in the spent fuels were systematically recovered (through reprocessing) and recycled in fast reactors, subject to technical developments and economical considerations. Undiscovered uranium resources are believed to amount to about 11.4 million tU [15]. Besides conventional resources, large resources of uranium exist. Phosphates are known to contain significant amounts of uranium, and industrial production has been performed in the USA and Belgium for the past twenty years. Another practically inexhaustible potential source of uranium is seawater (see Table 2.1).

#### *The perspective for the next twenty years*

However, despite these formidable resources, only highly localised deposits amenable to *in-situ* leaching technology with a uranium concentration of about 500 ppm or more or other deposits with grades above a few tenths of a percent, can be mined economically in today's market conditions, i.e. at prices  $<$ 40 USD/kgU. Uranium is produced using both conventional mining (open pit and underground) and unconventional production techniques. Unconventional techniques include *in-situ* leaching (ISL), phosphate by-product recovery<sup>7</sup> and heap leaching. ISL technology has a very low environmental impact but can be used only to extract uranium from suitable sandstone-type deposits. The distribution of production for the four technology types or material sources is shown in Table 2.2. Conventional mining and milling remains the dominant technology for producing uranium, accounting for nearly 80% of the uranium production in 1996 to 1998.

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7. The only three plants using phosphate by-product recovery (two in USA and one in Belgium, total production 495 tU/yr) suspended their uranium production in early 1999.

Table 2.1 Overview of uranium resources, production and projected capabilities [15]

| Resources (1 000 tU) reported in 1999  |              |           | Number of years of present nuclear electricity production <sup>8</sup> |               |               |               |               |
|--|--------------|-----------|--|---------------|---------------|---------------|---------------|
| Natural Uranium stocks   |              | 200       | 4  |               |               |               |               |
| HEU and Pu   |              | 600       | 12   |               |               |               |               |
| Known Conventional Resources   | <40 USD/kgU  | >1 254    |  |               |               |               |               |
|  | <80 USD/kgU  | 3 002     |  |               |               |               |               |
|  | <130 USD/kgU | 3 954     | 80   |               |               |               |               |
| Undiscovered Conventional Resources  | <80 USD/kgU  | 1 460     |  |               |               |               |               |
|  | <130 USD/kgU | 5 338     |  |               |               |               |               |
|  | Total        | 11 459    | 230  |               |               |               |               |
| Uranium in phosphates  |              | 22 000    | 440  |               |               |               |               |
| Uranium in seawater  |              | 4 200 000 | 80 000   |               |               |               |               |
| Production (tU) in 1998  |              |           |  |               |               |               |               |
| World  | 34 986       |           | OECD   |               | 19 088        |               |               |
| Projected production capabilities based on Known Conventional Resources (tU/y) |              |           | 1999   | 2000          | 2005          | 2010          | 2015          |
| Existing and committed   |              |           | 45 800 <sup>9</sup>  | 43 800        | 41 700        | 37 600        | 33 000        |
| <i>Total</i>   |              |           | <i>46 000</i>  | <i>45 750</i> | <i>61 200</i> | <i>64 800</i> | <i>55 000</i> |

Table 2.2 Distribution of uranium production by technology or source [15]

| Technology/<br>source | 1981 | 1994 | 1995 | 1996 | 1997 | 1998 |
|-----------------------|------|------|------|------|------|------|
| Open pit              | 68%  | 38%  | 37%  | 39%  | 49%  | 47%  |
| Underground           | 27%  | 41%  | 43%  | 40%  | 32%  | 34%  |
| ISL                   | 5%   | 14%  | 14%  | 13%  | 13%  | 14%  |
| Other*                |      | 7%   | 6%   | 8%   | 6%   | 5%   |

\* Phosphate by-product, heap and in-place leaching and mine water recovery.

In the next twenty years, low-cost uranium production from new projects is expected to be primarily from high-grade unconformity-type deposits, multi-mineral deposits and sandstone-type deposits amenable to ISL mining technology. It is expected that the use of conventional production technology will increase, with particular emphasis on underground mining, while ISL technology could maintain its relative share.

Today, primary production amounts only to 60% of the total world demand. Four additional (secondary) uranium resources make up the balance of supply and demand, i.e. uranium stockpiles, re-enrichment of depleted uranium, recycling of uranium and plutonium and the use of former weapons-grade material (HEU).

Since the early 1950s, Governments held important stockpiles of uranium as a result of their military programmes. Commercial stockpiles add to this secondary resource but a draw-down of these inventories has been experienced in the last few years. The main impacts of the inventory draw-down is to postpone the expansion of existing production projects or the development of committed, planned and prospective production centres, and to exert a downward pressure on uranium prices.

8. The number of years of present nuclear electricity production (in 2000: 2 540.5 TWhe [Nucleonics Week, February 8, 2001]), is calculated using a thermal efficiency of 35%, average load factor of 85%, and a ratio of natural to enriched uranium of  $8 \text{ kgU}_{\text{nat}}/\text{kgU}_{\text{enr}}$  (3.7%).

9. With a projected plant capacity utilisation of about 75%, existing and committed capability is about 74% of 1999 world uranium requirements.

Another potential source of uranium is the re-enrichment of depleted uranium<sup>10</sup> by using today's over-capacity of enrichment.<sup>11</sup> However, the economics of re-enrichment depend on the <sup>235</sup>U assay of the depleted uranium and the relationship between the price of uranium and the cost of enrichment services. In the Russian Federation, re-enrichment of depleted uranium for the production of low-enriched uranium (LEU) has taken place for several years.

Recycling of uranium and plutonium in thermal reactors provides a potential saving of 15-20% in natural uranium requirements. While the world-wide impact of U/Pu recycling on uranium demand remains small today, the savings potentially could contribute to a more sustainable use of the available resources. In this context, Japan has an aggressive plan for utilising MOX fuel in LWRs, because it has few domestic energy resources. In addition to the fabrication of MOX fuel in Europe, JNFL has started preparation of domestic fabrication of MOX fuel. However, within today's political and economic context, no major expansion of recycling is envisaged for the coming years in OECD Member countries.

The continued availability of new supplies from the conversion of ex-weapons material implies that production will be supplemented by large sales of LEU produced from HEU,<sup>12</sup> and from government stockpiles. This could apply a downward pressure on prices and thereby further delay expansion of the primary production capability.

Figure 2.3 shows the world's past and projected uranium production, requirements and prices. The projected production capabilities of existing and committed production centres<sup>13</sup> are expected to fall from 45 800 tU in 1999 to about 33 000 tU in 2015 due to the resorption of ex-military stocks and the closure of existing mines because of resource depletion or economic viability. Existing and committed capability at USD 40/kgU or less is likely to meet only about 40 to 60% of the requirements in 2015. The need for uranium will, however, continue as long as nuclear electricity generation continues, and new production will be needed in large quantities sometime in the period to supply a large share of requirements, once the excess supplies from past military efforts will be exhausted. In the longer term, resources recoverable at higher costs and additional supplies would be necessary to fill the potential production shortfall indicated by some of the projections. Overall world reactor-related uranium requirements are expected to evolve from about 60 488 tU in 1996 to levels ranging between 55 000 and 80 000 tU per year by the year 2015.

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10. The world-wide depleted uranium stock amounted to 1 180 000 tonnes U in 1999, with an additional annual arising of 51 400 tonnes U being expected up to about 2010 [15].
  11. The enrichment capacity in 2000 was reported to be 56 MSWU/year, compared with an annual demand of 35 MSWU/year. Not all of this enrichment capacity is economically competitive. An equivalent of around 7 000 tonnes/year of natural uranium is thought to be the limit of this source if this spare capacity would be utilised for enrichment of depleted uranium with an assay of 0.3%, drawing down to a new tails assay of 0.1%.
  12. The Russian and US Governments agreed to a programme to convert 500 tonnes HEU from weapons into LEU for civil reactors. These 500 tonnes HEU at 90% <sup>235</sup>U will be blended with 14 759 tonnes of LEU at 1.5% <sup>235</sup>U to obtain a product of 15 259 tonnes of LEU at 4.4% <sup>235</sup>U. This blending process would take place over a period of 20 years. By using depleted uranium and HEU to obtain the LEU at 1.5% <sup>235</sup>U, this agreement will replace 152 000 tonnes of natural uranium which could have been enriched instead to yield the 15 259 tonnes of LEU at 4.4% <sup>235</sup>U (with tails at 0.3% <sup>235</sup>U).
  13. Today's uranium production comes from about 63 mining centres, with concentrates produced at about 44 processing plants (mills). Employment associated with this uranium production accounted for about 53 700 person-years in 1998, declining from about 64 000 person-years in 1994. After decreasing for more than 10 years, exploration expenditures increased to about 130 million USD world-wide in 1998.

Uranium recovery from seawater has been studied in Japan as a supply option for the very long term or to provide supplies in the event of a very strong development of fission energy. Yet, at this stage of the study, it is difficult to predict the industrial application of uranium recovery from seawater. On a laboratory scale, experiments performed by JAERI, in which uranium is trapped on amidoxime adsorbent,<sup>14</sup> produced 1.7 gU/kg-adsorbent after 60 days of contact with seawater [16]. An economic assessment has been reported [4] indicating a possible cost for this process in a 1 000 t/year commercial plant of approximately USD 600/kgU. Ultimately, with an improved adsorbent, a recovery cost of about USD 220/kgU was reported to be feasible in the longer term compared to the present natural uranium price of about USD 30/kgU.

### *Processing of uranium ore*

Once the uranium ore has been mined, it is chemically treated to extract uranium salts for further processing and to achieve a volumetric reduction of the amount of material to be handled in the fuel cycle. The final product of this uranium ore treatment is uranium-concentrate, called “Yellow Cake”, containing about 70% uranium. The uranium ore treatment usually consists of four main process steps: ore treatment (milling, breaking), chemical attack (oxidation, leaching), uranium extraction (ion-exchange, solvent-extraction) and uranium recovery (precipitation, product washing, drying, packing). These processes mostly take place at facilities on the uranium mine site, or in facilities serving several nearby uranium mines, and the processing capacity is matched closely to the uranium production capacity served.

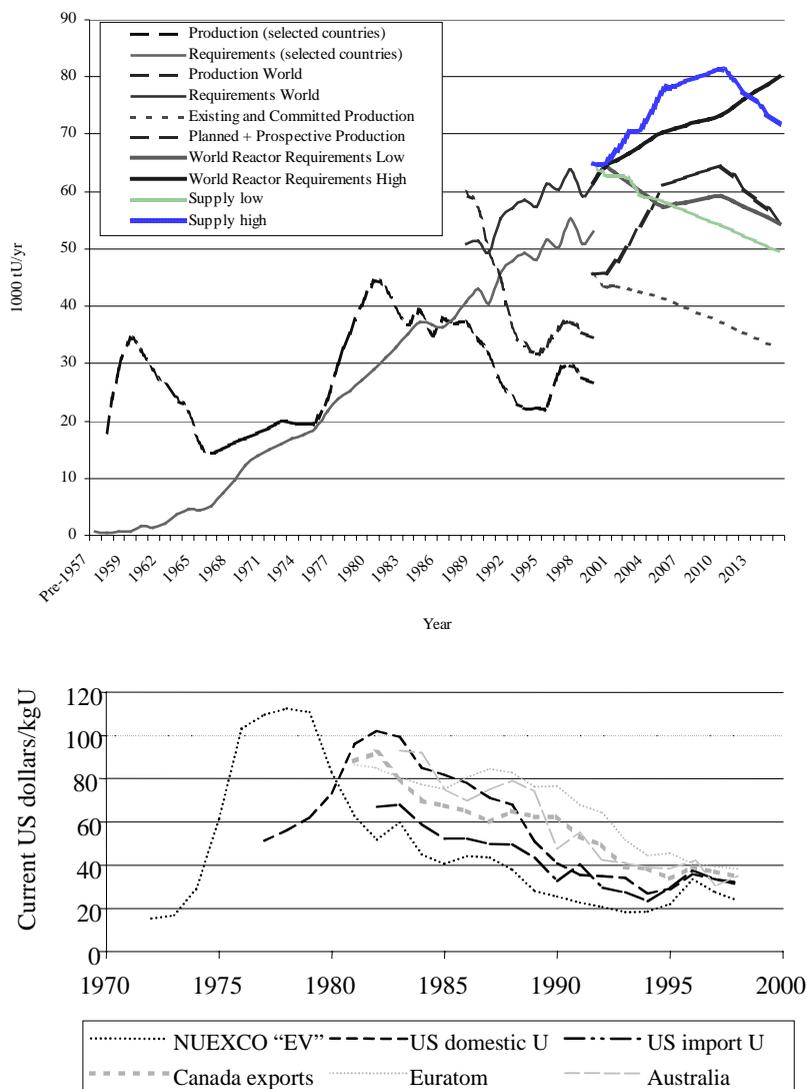
### *Uranium mining and mill tailing environmental impacts*

The environmental impact of the extraction and processing of uranium is mainly limited to the mining and mill tailings. The quantity of ore required to produce a tonne of uranium depends on the average grade of the ore and typically amounts to about 10 to 1 000 tonnes ore/tU (grade 10%-0.1% U). The higher-grade deposits now being developed will require a much lower rate of ore extraction but at the expense of special radiation protection measures for the workers. During open-pit mining operations, earth overburden above the ore-body and barren rock are produced. The quantity of waste rock removed depends on the deposits but is estimated to be around 5 000 tonnes per tU of annual production. The radiological impact is related mainly to the release of radon during mining<sup>15</sup> and especially from the mill tailings; this impact accounts for a collective dose of 0.8-1.0 manSv per GWe×year [17] while the global collective dose for the nuclear fuel cycle is calculated to be about 1.6-2.6 manSv per GWe×year. New mining techniques and environmental protection measures are developed in order to decrease the impact of mining and milling even further.<sup>16</sup> It should be noted that ISL mining does not produce tailings. Also, extraction of uranium from existing gold mine tailings does take place in South Africa, effectively reducing the environmental burden of these specific tailings and at the same time improving the gold extraction efficiency. In general, it should be remarked that the environment around uranium mines usually exhibits relatively high natural

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14. The adsorbent was prepared from non-woven material of polyethylene with the aid of radiation-induced co-grafting.
  15. The releases from uranium mining are, for the most part, similar to releases from other mineral mining. They are, in addition to typical releases, radon and radon progeny, radioactively contaminated water, dust, and others.
  16. The 1993 UNSCEAR report [17] mentioned a radon exhalation rate from bare tailings ranging between 10 and 300 Bq.m<sup>-2</sup>.s. In a recent study by SENES Consultants [18] and others [19,20], information was gathered from eight major uranium production facilities, indicating releases estimated to range from 0 to 7 Bq.m<sup>-2</sup>.s.

backgrounds of radioactivity due to the presence of above-average concentrations of uranium in the earth (which, obviously, is the reason it was chosen as a mine site). This makes it rather difficult to distinguish between the releases and exposures caused by the mining from those due to the naturally present background radioactivity.

Figure 2.3 Uranium production, requirements and prices<sup>17</sup> [15]



17. The production and requirements from "selected countries" excludes Bulgaria, China, Cuba, Czech Republic, GDR, Hungary, Kazakhstan, Mongolia, Romania, Russian Federation, Slovenia, Tajikistan, Ukraine, USSR, Uzbekistan and Yugoslavia. The low supply curve corresponds to the existing and committed production and the addition of the secondary uranium resources; the high supply curve corresponds to the addition of planned and prospective production and the secondary resources.

### 2.2.1.2 Thorium

The use of thorium has known little development within OECD Member countries. Thorium, which is essentially 100%  $^{232}\text{Th}$ , is three times more abundant<sup>18</sup> in the earth's crust than uranium and occurs in association with uranium and rare earth elements in diverse rock types. Estimates indicate that the resources are somewhat larger than for uranium (see Table 2.3) [21, 22]. However, owing to the fact that thorium resources have not been surveyed as intensively as uranium, and because thorium deposits are expected to be well dispersed and relatively dilute, any assessment of the quantities of "reasonably assured" and "estimated additional" resources, as well as the amounts of these that might be economically recoverable, should be considered as being conservatively on the low side. Among them, monazite has the most potential due to the existence of large resources containing 5 to 10% thorium. Monazite is found in beach sands in India, Australia, Brazil and Egypt and is also present in some stream placer deposits in Idaho and in the south-eastern part of the USA but at much lower concentrations (up to few tenths of a per cent). Monazite is exploited from placers for production of rare earth elements, Zr and Ti, with most of the thorium currently being left in the residues. The world's monazite concentrate production amounts to about 8 500 t/year, including about 200 t/year of thorium.

During 1950-1970, when uranium was considered a rare commodity, a number of options for energy production with thorium were investigated in the USA, the former USSR, Europe and Asia. Work on thorium cycles focused mainly on the potential for enhancing uranium use in thermal systems. Since the beginning of the 90s, additional interest in thorium fuel arose for LWRs and accelerator-driven systems (ADS) because it might generate less long-lived minor actinides than the traditional uranium fuel cycle (see Chapter 4).

Table 2.3 **Estimated world thorium resources in 1992**  
(market economy countries) in  $10^3$  t [21,22]

| Countries          | RAR*         | EAR**             | Total        | %          |
|--------------------|--------------|-------------------|--------------|------------|
| <b>Europe</b>      | 566          | 724               | 1 230        | 31         |
| Greenland          | 54           | 32                | 86           | 2.2        |
| Norway             | 132          | 132               | 264          | 6.6        |
| Turkey             | 380          | 500 <sup>19</sup> | 880          | 22.1       |
|                    |              |                   |              |            |
| <b>America</b>     | 790          | 1 125             | 1 911        | 48         |
| Brazil             | 606          | 700               | 1 306        | 32.8       |
| Canada             | 45           | 128               | 173          | 4.4        |
| United States      | 137          | 295               | 432          | 11         |
|                    |              |                   |              |            |
| <b>Africa</b>      | 36           | 309               | 439          | 11         |
| Egypt              | 15           | 280               | 295          | 7.4        |
| Niger              | ?            | ?                 | 29           | 0.7        |
| South Africa       | 18           | ?                 | 115          | 2.9        |
|                    |              |                   |              |            |
| <b>Asia</b>        | 343          | 30                | 403          | 10         |
| India              | 319          |                   | 391          | 9.8        |
|                    |              |                   |              |            |
| <b>World total</b> | <b>1 754</b> | <b>2 188</b>      | <b>3 983</b> | <b>100</b> |

\* RAR: reasonably assured resources

\*\* EAR: estimated additional resources

18. Thorium is as abundant as tin in the earth's crust. Uranium is more abundant than silver.

19. This EAR Th-resource value is currently investigated by KAEC by performing drillings and preparation of maps. Preliminary data indicate low Th-content yields (approx. 0.3%) for the sites investigated. These investigations are covered by "The Project for Development and Projection of Mapping Turkish Th-Resources" sponsored by TAEA (Project nr. TPD98K120180).

The environmental impact of thorium extraction is, as in the case of uranium, related to the mining and ore treatment processes. In the case of a beach placer mined by using dredging techniques, the amount of beach sand dredged depends on the concentration of heavy minerals in the sand. In the Indian Travancore beach placer, the amount of beach sand processed varies from 1 400 to 11 400 t/tTh metal produced.<sup>20</sup> On the other hand, in the case of Florida beach placers, which contain about 3% heavy minerals in which 1% is thorium, the amount of sand to be processed is much larger, of the order of 100 000 t/tTh metal produced. The sand usually is left on the mining site and does not pose any major environmental problem, although removing the monazite reduces the natural external radiation level in the immediate environment.

## 2.2.2 Conversion of the raw materials into primary fuels

While the result of the first steps of the fuel cycle is “yellow cake”, i.e.  $U_3O_8$  concentrate, the second main step yields a high purity uranium product. The chemical and physical form of the converted product depends on the intended further use of the uranium, i.e. fuel type, and the technologies used for enrichment of the uranium. One could, in principle, distinguish between conversion to metallic or oxide (non-enriched) uranium for use in Magnox and CANDU reactors and the conversion to uranium hexafluoride ( $UF_6$ ) intended for enrichment with today’s enrichment processes. Future developments in enrichment technology (e.g. laser enrichment) could change the needed chemical and physical form of the converted product.

### 2.2.2.1 Conversion from $U_3O_8$ to $UF_6$

The conversion from yellow cake to  $UF_6$  is achieved by dissolving the uranium concentrate in nitric acid, filtering and purifying the solution with chemical solvents. The resulting uranyl nitrate (>99.95% pure) is then reconverted to uranium oxide and this, in turn, is converted to readily vaporisable uranium hexafluoride<sup>21</sup> that is used in the enrichment process. If enrichment is not required, for example for CANDU-fuel, then uranium dioxide is produced from the uranyl nitrate and shipped directly to a fuel fabrication plant. Conversion plants are highly specialised chemical facilities and, because they handle some very aggressive chemicals (F, HF), they are subject to rigorous operational conditions. They do not, however, produce significant amounts of radioactive effluents (principally containing natural uranium  $\alpha$ -activity). Re-conversion of depleted uranium hexafluoride to  $U_3O_8$  currently is practised in France and planned in the USA in order to cope with the growing stock of depleted uranium hexafluoride, taking the view that storage as stable uranium oxide is more appropriate for the very long term.

Table 2.4 gives an overview of the major commercial conversion plants and processes in use today or planned in the near future.<sup>22</sup> Some other conversion plants are mentioned, that are mainly annexes to the commercial plants and serving specific conversion needs.

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20. The original sand contains 20 to 40 wt% of heavy minerals, among which 0.5 to 2% is monazite and the thorium content in this monazite is as high as 8.8%.
  21. At ambient temperature,  $UF_6$  is a colourless, high molecular weight (352) solid with a significant but less than atmospheric vapour pressure. It is readily transformed into a gas at atmospheric pressure by raising its temperature above 56.4°C and into a liquid by increasing the pressure above 1.5 atmospheres and the temperature above 64°C. All three phases, solid, liquid and gas, coexist at 64°C.
  22. Argentina, Brazil, China, Korea, Pakistan and Japan have conversion plants as well.

Table 2.4 Conversion plants and processes [23]

| Land               | Place               | Operator  | Capacity (tU/year)   | Conversion process                              | First operation |
|--------------------|---------------------|-----------|----------------------|---|-----------------|
| Canada             | Blind River         | Cameco    | 18 000               | $U_3O_8 \Rightarrow UO_3$ , wet, UNH-process    | 1983            |
|                    | Port Hope           | Cameco    | 2 000                | $UF_4 \Rightarrow U$ -metal                     | 1985            |
|                    | Port Hope           | Cameco    | 10 500               | $UO_3 \Rightarrow UF_6$                         | 1984            |
|                    | Port Hope           | Cameco    | 3 800                | $UO_3 \Rightarrow UO_2$                         | 1980            |
| France             | Malvési             | Comurhex  | 14 000               | $U_3O_8 \Rightarrow UO_4$ , wet, ADU-process    | 1959            |
|                    | Pierrelatte         | Comurhex  | 350                  | $UNH \Rightarrow UF_6$                          | 1976            |
|                    | Pierrelatte         | Comurhex  | 14 000               | $UF_4 \Rightarrow UF_6$                         | 1961            |
|                    | TU 2                | COGEMA    | 350                  | Depleted $U_3O_8 \Rightarrow UO_2$ (MOX top-up) | 1988            |
|                    | TU 2 Reprocessing   | COGEMA    | 1 200                | $UNH \Rightarrow U_3O_8$                        | 1988            |
|                    | TU 5 Reprocessing   | COGEMA    | 2 000                | $UNH \Rightarrow U_3O_8$                        | 1995            |
|                    | W Defluorination    | COGEMA    | 20 000               | $UF_6 \Rightarrow U_3O_8$                       | 1984            |
| Russian Federation | Angarsk             | Minatom   | 20 000 <sup>23</sup> | $U_3O_8 \Rightarrow UF_6$                       | 1954            |
|                    | Ekaterinburg        | Minatom   | 4 000                | $U_3O_8 \Rightarrow UF_6$                       | 1949            |
| UK                 | Springfields Line 4 | BNFL      | 6 000                | $U_3O_8 \Rightarrow UF_6$ , wet, UNH-process    | 1974/94         |
| USA                | Metropolis          | ConverDyn | 12 700               | $U_3O_8 \Rightarrow UF_6$ , dry                 | 1959            |

The Russian conversion plants appear to be tightly integrated to their enrichment facilities, and do not export their product on the Western markets. Although the  $UF_6$  conversion capacities in the Western countries appear sufficient in the short-term to convert the primary uranium actually produced, there are uncertainties about adequacy of future conversion capacities when the BNFL plant will be closed (in 2006) and uranium primary production will increase to compensate for disappearance of secondary sources.

#### 2.2.2.2 Enrichment

Uranium enrichment involves the partial separation of natural uranium into its two main isotopes  $^{235}U$  and  $^{238}U$ , yielding an enriched fraction containing more  $^{235}U$  and a depleted fraction, called “tails”, containing less than the natural (0.711 wt%) concentration of  $^{235}U$ .

The early history of uranium enrichment was associated with the production of nearly pure  $^{235}U$  for nuclear weapons material, and many separation methods were considered. These included gaseous diffusion, centrifuges, aerodynamic processes, chemical and photochemical methods in addition to a number of other techniques such as electromagnetic separators and distillation. Of all these processes, the most feasible technique seemed to be electromagnetic separation, gaseous diffusion and gas centrifugation. The electromagnetic method, i.e. using calutrons, was deployed during the early period of fissile material production for the US weapons programme, but has not been used extensively for uranium enrichment, its use currently being confined to the supply of other stable isotopes. The main difficulty of this technique for commercial production of enriched uranium lies in the small throughputs that can be achieved.

The main route for uranium enrichment after World War II was through gaseous diffusion, and although there was interest, going back to the beginning of the 1940s, in the use of centrifuges, the technical and engineering problems encountered were such that centrifugation processes only became

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23. The amount which can be supplied to customers in the Western World is currently limited by the US Suspension Agreement and by Euratom policy to about 7 000 tU per year.

commercial in the seventies. Today, gas diffusion and centrifuge technologies are used by the different enrichment companies world-wide (see Table 2.5).

Table 2.5 **Today's commercial enrichment facilities**<sup>24</sup> [23]

| Land               | Place                       | Operator | Capacity<br>(1 000 SWU/year) | Enrichment process         | First operation |
|--------------------|-----------------------------|----------|------------------------------|----------------------------|-----------------|
| China              | Lanzhou                     | CNNC     | 450                          | Gaseous diffusion          | 1980            |
|                    | Shaanxi                     |          | 250                          | Centrifuge                 | 1997            |
|                    |                             |          | 200                          |                            | Being installed |
| France             | Tricastin<br>(George Besse) | Eurodif  | 10 800                       | Gaseous diffusion          | 1979            |
| Germany            | Gronau                      | Urenco   | 1 300                        | Centrifuge                 | 1985            |
| Japan              | Rokkasho-Mura 1             | JNFL     | 600                          | Centrifuge                 | 1992            |
|                    | Rokkasho-Mura 2             | JNFL     | 450                          | Centrifuge                 | 1997            |
| Netherlands        | Almelo                      | Urenco   | 1 500                        | Centrifuge                 | 1973            |
| Russian Federation | Angarsk                     | Minatom  | 1 400                        | Centrifuge                 | 1954            |
|                    | Ekaterinburg                | Minatom  | 10 000                       | Centrifuge                 | 1949            |
|                    | Krasnoyarsk                 | Minatom  | 2 900                        | Centrifuge                 | 1964            |
|                    | Seversk                     | Minatom  | 5 700                        | Centrifuge (reprocessed U) | 1950            |
| UK                 | Capenhurst                  | Urenco   | 2 000                        | Centrifuge                 | 1976            |
| US                 | Paducah                     | DOE/USEC | 11 300                       | Gaseous diffusion          | 1954            |
|                    | Portsmouth <sup>25</sup>    | DOE/USEC | 7 400                        | Gaseous diffusion          | 1956            |
| <i>Total</i>       |                             |          | <i>56 250</i>                |                            |                 |

### *Enrichment market*

The market for enrichment services is expected to show little growth over the next two decades, as there is an over-capacity (approximately 15-20 million SWU). It is estimated that world demand will increase to about 37 million SWU by 2005, driven primarily by increased demand in Asia and will remain constant until 2010.

Over the recent years, utility enrichment bid requests have been met with very aggressive competition from primary enrichers seeking to build market share in an over-supplied market. Contract prices declined, with offers in the range of USD 78-80/SWU (NUKEM, 2000). Nominal prices for enrichment in the coming years are expected to be in the 80-120 USD/SWU range.

Table 2.5 also indicates that replacement of today's facilities will become an issue in the coming decade. As of 2005, 90% of the enrichment industry's capacity will be in equipment which will be more than 15 years old; about 70% will be more than 25 years old. This applies to the operating equipment, much of which is located in plant facilities that are substantially older. The question of replacement of the gaseous diffusion plants by high-performance gas centrifuges or laser isotope separation technology will have to be addressed soon [24]. However, laser isotope separation development has been scaled-down drastically and even stopped in the USA and France. New enrichment plants in the medium term in Japan, USA and probably France will, therefore, likely be based on centrifuge technology.

Laser enrichment by AVLIS was dropped by USEC on technical reasons and short-term economics, but could make a come back if reprocessing is further developed, as only laser enrichment

24. Other centrifuge facilities are planned in Brazil and China.

25. Will cease operation in June 2001 and will be put in stand-by condition (www.usec.com).

allows sustainable recycling of reprocessed uranium (separating the minor isotopes  $^{232}\text{U}$ ,  $^{234}\text{U}$  and  $^{236}\text{U}$  from  $^{235}\text{U}$ ). USEC is still pursuing on a laboratory scale laser enrichment technology with SILEX. In the longer term, and the more so if there is a nuclear renaissance, future enrichment facilities will be needed and these will have to rely upon advanced processes.

There is ongoing research and development in order to improve the current enrichment facilities. JNC (Japan) has conducted projects to develop centrifuge technology and operated pilot and demonstration plants at Ningyo-Toge which were closed in March 2000. The technology developed was transferred to the construction and operation of the Rokkasho-Mura plants. In order to improve the economy, JNC has developed high-performance centrifuges with new composite materials. Also exploiting the advantage of having low energy requirements and sunk capital early in advanced technology plants, Urenco continues its improvement programme of gas centrifuge technology, focusing on increased separation work output per machine, decreased construction costs for the centrifuges and increased reliability and lifetime of the centrifuges.

The mass balance of enrichment facilities indicates that only very small quantities of uranium are vented from the process and auxiliary systems of gaseous diffusion plants while radioactive discharges from centrifuge enrichment facilities are even smaller. For instance, atmospheric releases from EURODIF in 1997 were 3.3 kg uranium, with a total  $\alpha$ -activity of 0.16 GBq, and liquid releases were only 0.0094 GBq for uranium [24].

The enrichment of uranium produces depleted uranium (DU) as a by-product. For a typical LWR, between 6 and 8 tons of DU are produced per ton of fresh fuel. Different countries have adopted different management strategies for this material. France is converting the DU to a stable oxide for long-term storage and possible eventual reuse as fuel in breeder reactors or for extraction of additional fissile  $^{235}\text{U}$ . No decisions have been made in the United States on long-term management of this by-product; however, the USA Nuclear Regulatory Commission has stated that some type of deep disposal is required.

### 2.2.2.3 Fuel design and fabrication

About 20% of the total costs for nuclear electricity generation relate to the fuel cycle. In turn, about 15% of the fuel cycle costs, or only about 3% of the total electricity costs, relate to fuel fabrication (see Figure 2.2). However, in spite of this small share in the total costs, the influence of fuel design and fabrication on the overall economy of nuclear power production is considerable. Indeed, the fuel assembly represents the connecting link between the fuel cycle and the nuclear power plant. Improvement of fuel utilisation not only reduces the fuel cycle costs, but improved operational flexibility and reliability of the fuel assembly may lead also to improvement of the plant capacity factor and, thereby, to a reduction of the non-fuel shares of the total electricity costs. One should also remark that short-term minimisation of fuel cycle costs, e.g. with respect to cutting back individual process prices, although desirable, may not always lead to minimal power generation costs. Any improvement in fuel and its utilisation has to be seen as an integral part of an overall fuel cycle strategy.

Today's fuel vendors (see Table 2.6) are active in an extremely competitive market where fuel fabrication capacity is well in excess of both current and anticipated future requirements. On a world-wide basis, LWR fuel manufacturing capacity appears to be nearly 50% above annual requirements. In the light of this situation, fuel fabricators continue to innovate in their manufacturing facilities in order to lower costs. Due to consolidations between fuel vendors, excess capacity is likely to be reduced in the future.

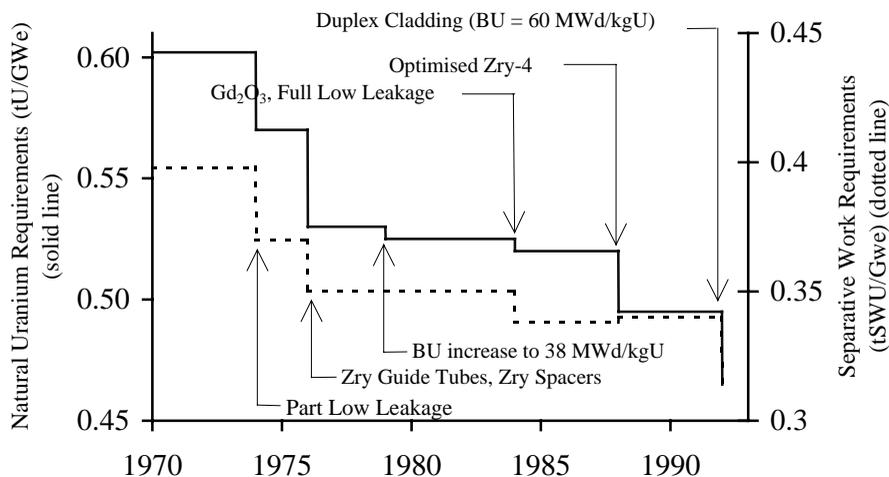
With respect to core reload design, changing reload patterns in the reactor core is receiving stronger emphasis by the reactor operators. While nuclear power plants likely will continue to be

operated mainly to meet base load demand, the timing of outages will be defined increasingly by market conditions, influenced by the marginal generation costs of nuclear versus other available sources at the time. The outcome of this will be some variations in fuel cycle lengths and, thus, differing needs in fuel enrichment levels, numbers of fuel assemblies and other factors in the core reload designs. As a result, core designs and the corresponding fuel supply will have to be more flexible.

Technical targets in achieving the objective of improving fuel performance aim at increasing the discharge burn-up and at improving fuel performance by best utilisation of margins and other measures, while maintaining an excellent reliability of the fuel assemblies. Therefore, next to development focused on the performance of the fuel in the reactor, a continuous quality assurance programme is ongoing with respect to fuel material and fuel element fabrication.

Depending on the special boundary conditions in different countries, about half of the fuel cycle costs can be influenced by improvement of the fuel utilisation and half by burn-up increase. In order to optimise fuel utilisation in the reactor, i.e. to generate the maximum amount of power using the minimum amount of uranium, one should consider the basic neutronic effects – moderation, absorption and leakage – in a reactor core. Optimisation of the fuel/moderator distribution, reduction of parasitic absorption and reduction of radial and axial neutron leakage have been main objectives in the area of fuel utilisation. Figure 2.4 shows an example<sup>26</sup> of the design advances and their contribution to improving fuel utilisation in the past [25]. These developments resulted in a total saving of more than 25% on natural uranium requirements per unit of electricity produced, as well as significantly improving the fuel cycle costs.

Figure 2.4 **Impact of fuel design and fabrication on enrichment and U requirements** [25]



26. For the case of Siemens PWR fuel assembly.

Table 2.6 Commercial fuel fabrication plants [23]

| Country             | Place          | Operator                           | Fuel-type               | Capacity (tHM/year) | First operation |
|---------------------|----------------|------------------------------------|-------------------------|---------------------|-----------------|
| <b>LWR</b>          |                |                                    |                         |                     |                 |
| Belgium             | Dessel         | FBFC                               | LWR                     | 500                 | 1961            |
| Brazil              | Resende        | INB                                | LWR                     | 100                 | 1982            |
| China               | Yibin          | CNNC                               | LWR                     | 100                 | 1993            |
| France              | Romans         | FBFC                               | LWR                     | 800                 | 1979            |
| Germany             | Lingen         | Advanced Nuclear Fuels GmbH        | LWR                     | 650                 | 1979            |
| India               | Hyderabad      | DAE                                | UO <sub>2</sub> pellets | 300                 | 1998            |
| India               | Hyderabad      | DAE                                | BWR                     | 25                  | 1974            |
| Japan               | Tokai-Mura     | MNF                                | PWR                     | 440                 | 1972            |
|                     | Kumatori-machi | NFI                                | PWR                     | 284                 | 1972            |
|                     | Tokai-Mura     | NFI                                | BWR                     | 200                 | 1980            |
|                     | Kurihama       | JNF                                | BWR                     | 750                 | 1970            |
| Korea               | Yusung         | KNFC                               | PWR                     | 400                 | 1989            |
| Russian Federation  | Elektrostal    | Mashino Stroitelny                 | UO <sub>2</sub> pellets | 800                 | 1996            |
|                     | Elektrostal    | Mashino Stroitelny                 | LWR (VVER)              | 620                 |                 |
|                     | Novosibirsk    | TVEL                               | LWR (VVER)              | 1 000               | 1949            |
| Spain               | Juzbado        | ENUSA                              | LWR                     | 300                 | 1985            |
| Sweden              | Västerås       | ABB                                | LWR                     | 600                 | 1971            |
| UK                  | Springfields   | BNFL                               | LWR                     | 330                 | 1996            |
| USA                 | Hematite       | ABB-CE                             | LWR                     | 450                 | 1986            |
|                     | Columbia       | Westinghouse                       | PWR                     | 1 150               | 1986            |
|                     | Lynchburg      | FC Fuels                           | PWR                     | 400                 | 1982            |
|                     | Richland       | Framatome ANP, Inc.                | LWR                     | 700                 | 1970            |
|                     | Wilmington     | GE Nuclear Energy                  | BWR                     | 1 200               | 1982            |
| <b>PHWR</b>         |                |                                    |                         |                     |                 |
| Argentina           | Ezeiza         | CNEA                               | PHWR                    | 160                 | 1982            |
| Canada              | Toronto        | GE Canada Inc.                     | Pellets                 | 1 300               | 1967            |
|                     | Peterborough   | GE Canada Inc.                     | PHWR                    | 1 200               | 1956            |
|                     | Port Hope      | Zircotec Precision Industries Inc. | PHWR                    | 1 500               | 1964            |
| India               | Hyderabad      | DAE                                | PHWR                    | 300                 | 1974            |
|                     | Trombay        | DAE                                | PHWR                    | 135                 | 1982            |
| Korea               | Kaeri          | KAERI                              | PHWR                    | 400                 | 1987            |
| <b>Others (AGR)</b> |                |                                    |                         |                     |                 |
| UK                  | Springfields   | BNFL                               | Magnox (GCR)            | 1 300               | 1960            |
|                     | Springfields   | BNFL                               | UO <sub>2</sub> , AGR   | 290                 | 1996            |
| <b>FBR</b>          |                |                                    |                         |                     |                 |
| France              | Veurey-Voroise | CISN                               | FBR                     | 150                 | 1960            |
| Russian Federation  | Elektrostal    | Mashino Stroitelny                 | FBR                     | 50                  |                 |
| <b>MOX Fuel</b>     |                |                                    |                         |                     |                 |
| Belgium             | Dessel         | Belgonucléaire                     | LWR                     | 40                  | 1973            |
| France              | Cadarache      | COGEMA                             | LWR, FBR                | 40                  | 1961            |
|                     | Marcoule-Melox | COGEMA                             | LWR                     | 100                 | 1995            |
| UK                  | Sellafield SMP | BNFL                               | LWR                     | 120                 | 200?            |
|                     | Sellafield MDF |                                    | LWR                     | 8                   | 1993            |
| Japan               | Tokai-Mura     | JNC                                | ATR                     | 10                  | 1972            |
|                     |                |                                    | FBR                     | 5                   | 1988            |
|                     | Rokkasho       | JNFL                               | LWR                     | 130                 | 200?            |
| Russian Federation  | Chelyabinsk    | Minatom                            | FBR                     | 60                  |                 |

Increasing the discharge burn-up reduces the demand for fuel assemblies and natural uranium for a given amount of energy production, influencing in this way both fuel supply and disposal volumes. Depending on the related boundary conditions in the different countries, this results in considerable economic savings. In the past, increase of discharge burn-up was the predominant effect driving the fuel cycle economy in many countries, and this still is a continuing trend today. Material development as well as material and prototype test programmes performed in advance of commercial deployment serve to verify knowledge of material properties and behaviour, and form a sound basis for successful further burn-up increase. However, most reactors are licensed for UOX fuel with fuel assembly average burn-ups limited to about 50-55 GWd/tHM. Other limits to increased fuel burn-up relate to regulatory limits of 5% <sup>235</sup>U on enrichment levels for transportation, storage and fuel assembly fabrication facilities as well as most often for enrichment facilities. Overcoming these regulatory limits, and providing the necessary technical solutions, to allow batch average burn-up to be raised to about 65-70 GWd/tHM would require investment of significant economic resources and would take considerable time.

Mechanical design of fuel rods using statistical methods has been gaining steadily in importance, especially as fuel assemblies now are subject to more stringent service requirements (e.g. in connection with higher enrichments and reactor up-ratings). These design methods satisfy the requirements that all available information be utilised in design analysis and verification, including the assessment of margins to design limits. Compared to deterministic approaches employed to date, fuel rod design using statistical methods provides numerous advantages, such as:

- Consistent treatment of code uncertainties and distributions of fabrication data.
- Realistic assessment of existing margins to design limits.
- Close co-ordination of fuel rod design and in-core fuel management, allowing optimum utilisation of design margins.
- Consistent description of rod behaviour during normal operation, at the onset of postulated accidents, and in dry spent fuel storage.

#### *2.2.2.3.1 MOX fuel fabrication*

Since the earliest days of the commercial utilisation of nuclear power, it was recognised that plutonium arising from reprocessing of spent fuel was used most effectively in FBRs. In the 1960s, interest in plutonium recycling in LWRs grew steadily as plutonium surpluses appeared unavoidable. As a result, several countries started activities on mixed uranium-plutonium oxide (MOX) utilisation in LWRs: US and Belgium (since 1950s), Germany, France, Italy, Japan, Sweden, Switzerland and UK. In the same period, MOX, up to then investigated alongside uranium-plutonium alloys, appeared also to be an adequate fuel form for the use of plutonium in FBRs. The development of mixed uranium-plutonium oxide fuel fabrication was therefore supported by FBR R&D activities, in particular in the US, France, Germany, Belgium, Japan and the UK. Pu recycling in current PWRs is an industrial reality in Belgium, France, Germany and Switzerland. The understanding of the behaviour of plutonium fuel has grown significantly since the launch of the first R&D programmes on LWR and FR MOX fuels. Plutonium oxide physical and neutronic behaviour is well known, and its modelling is now available as well as its experimental validation. Up to now, more than 750 tHM MOX fuel (more than 2 000 fuel assemblies) have been loaded in 29 PWRs and in 2 BWRs in Europe, corresponding to the recycling of about 35 t of plutonium. Currently, MOX fuel fabrication capacities in OECD countries represent a flow of 190 tHM per year. This level corresponds to some 10 to 12 tPu used in MOX fuel per year.

Today's MOX industry has become a mature business where parity between UOX and MOX fuel is requested by the utilities embarked in a recycling strategy. Developments currently pursued focus on smoothing fuel cycle management by increasing burn-up of the MOX fuels, the possibility of load-following with MOX fuelled reactors (already demonstrated and licensed in France) and the possibility of increased MOX fraction (100% MOX cores) resulting in a net reduction of the Pu inventory (-15 kg/tHM per cycle in a 900 MWe LWR with a 100% MOX core).

Previously, examination of irradiated fuel rods had shown that MOX fuel behaved similarly to  $\text{UO}_2$  with respect to water corrosion and deformation, was less sensitive than  $\text{UO}_2$  to PCI (pellet-cladding interaction), but was less favourable for gas releases because a high gas release was caused by poor homogeneity of the fuel microstructure. Recently, fabrication technology has been improved and the gas release behaviour of MOX has become almost the same as for  $\text{UO}_2$ . Experimental, analytical and irradiation programmes are continuing to qualify MOX fuel for higher burn-up by increasing the Pu content in the fuel. In addition, the progressive degradation of the Pu isotopic mix, resulting from the reprocessing of higher discharge burn-up  $\text{UO}_2$  fuel, will also require increasing the Pu content in the next MOX fuel generations. Such increases may have consequences on fabrication plants (MELOX is licensed up to 12% Pu content) and on reactor core safety parameters (void effect, etc.).

MOX fuel fabrication for LWR use is today mainly based on the MIMAS process (Belgonucléaire, COGEMA) and on the Short Binderless Route (BNFL). In Japan, MOX fuels for core physics and irradiation testing have been manufactured in the JNC plant since 1966. The JNC MOX fabrication plant has been operating since 1972 (10 tMOX/y) for the advanced thermal reactor ATR-“Fugen”, with nearly 130 t of fuel (about 750 fuel assemblies) having been produced by 1999, equivalent to about 1.8 tHM of plutonium. A microwave heating method for the co-conversion of plutonium-uranium nitrate to MOX powder was developed by JNC in order to obtain MOX powder directly from the product of the reprocessing process. This method improves the non-proliferation character by avoiding the existence of pure  $\text{PuO}_2$  anywhere in the process. JNFL decided in November 2000 on a new LWR-MOX fuel fabrication plant with a capacity of 130 tHM/y which will be constructed at Rokkasho-Mura and will be supplied with reprocessed fuel from the Rokkasho Reprocessing Plant. In Russia, similar to the programmes pursued in the OECD countries, plutonium has been considered as a nuclear fuel since the second half of the 1950s. Two principal technologies are being developed to process the plutonium (mainly low burn-up) into mixed uranium-plutonium fuel: pelletising of U-Pu powders and vibrocompacting<sup>27</sup> which directly uses the acid solution from reprocessing and which could have an economic merit compared to the classic pelletising route. These technologies are implemented at Mayak (Chelyabinsk) and at RIAR (Dimitrovgrad), respectively.

A short-term issue is that MOX use impacts the final waste management also. The increased accumulation of minor actinides (especially  $^{241}\text{Am}$  and  $^{244}\text{Cm}$ ) and the higher and longer-lasting residual heat in the fuel calls for an adapted waste management policy (see Section 2.2.3). In the longer term, multi-recycling of MOX fuel is a possible option, which is predicted not only to reduce the inventory of separated plutonium, but also to prevent an increasing fraction of minor actinides in the discharged fuel. Improvements in this balance can be made by the use of over-moderated MOX cores, resulting in the ratio of net minor actinide production over plutonium depletion being reduced and remaining small after successive recycling. These improvements demand, however, changes to the core and as such could be envisaged only as possibilities in advanced reactor concepts in the next decades (see Chapter 4). It should be mentioned that no specific problems emerge in the modelling and safety evaluation of the use of MOX in current and future LWRs [26].

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27. This production technique is of high interest for future development of Am+Cm fuels or targets in advanced nuclear fuel cycles (see Chapter 4).

Disposition of ex-weapons plutonium is a matter of concern for the political world, and this concern can be addressed by the nuclear industry. Several processes for conversion, MOX fabrication and MOX burning are currently proposed under international co-operation. The USA, adopting a dual track (MOX use and immobilisation) approach for Pu disposition, proposes to apply a dry hydride-based process to convert the plutonium into oxide for MOX source material. In addition, the USA is seeking an alternative using the CANDU reactors. Russia is pursuing a trilateral co-operation with France and Germany. The joint French-Russian studies under the AIDA-MOX Program concluded in favour of a process for MOX fuel fabrication based on the COCA and MIMAS processes after metallic plutonium is converted into  $\text{PuO}_2$  through  $\text{HNO}_3$ -HF dissolution and Pu oxalate precipitation. France, Germany and Russia jointly proposed the construction of a pilot scale MOX plant in Russia with a yearly capacity of 1.3 tPu, providing enough MOX fuel for a hybrid core BN-600 fast reactor and up to 7 VVER-1 000 reactors, including the 4 Balakovo units. Another option for Russia is to dispose of its ex-weapon plutonium as MOX fuel for the fast reactor BN-600. The core of BN-600 would have to be converted first to partial core and thereafter to full core MOX loading. Vibro-pack technology developed by Russia would be adopted for fuel fabrication. Japan is collaborating on core/fuel design, safety analysis, irradiation test and upgrading of the fuel fabrication facility.

#### 2.2.2.3.2 Use of reprocessed uranium

The nuclear industry has in place the facilities that are needed to recycle reprocessed uranium (REPU) from LWR on a semi-industrial scale [27]. This includes chemical conversion of REPU, enrichment, fuel fabrication and transport as well as reactor irradiation of REPU-based fuel. In Japan, JNC carried out experiments on the REPU conversion technology at the Ningyo-Toge conversion facility. The amount of REPU converted to  $\text{UF}_6$  for re-enrichment reached 336 tU. Recycling of REPU from LWR fuel is now demonstrated, albeit limited to a fraction of the available material and to only a few reactors.<sup>28</sup> The enrichment of REPU is today based on centrifuge technology and is currently performed in Russia and in the Netherlands. Altogether, the reprocessing plants have delivered more than 12 000 tonnes of REPU, and this amount will grow in the years to come, at a pace comparable to the output of a large uranium mine. More recently, an alternative way of REPU reuse has emerged which relies on blending REPU with highly-enriched uranium resulting in a reduced  $^{236}\text{U}$ -content in the enriched REPU fuel, thus reducing neutron absorption and improving the economy of REPU recycling. If REPU is considered as a by-product from reprocessing, the economics realised in replacing natural uranium by REPU compensate for the extra expenditures related to storage, conversion, enrichment and fabrication depending on the source material. The competitiveness of REPU is strongly related to the cost of reprocessing and the price of natural uranium, but also to its isotopic composition that varies from batch to batch. The future of REPU is anticipated to be governed by economic and strategic, e.g. conservation, considerations. Considering the savings that can be made by recycling REPU of relatively good specifications, despite an extremely depressed natural uranium market, it is possible that in the coming years those utilities that have decided to reprocess will also recover and recycle uranium. It should also be mentioned that the use of REPU would benefit from laser isotope enrichment technologies as this could more selectively re-enrich the  $^{235}\text{U}$ -content and thus improve the intrinsic value of the recycled uranium for further use in LWRs. Added to the direct economic benefit, considerations such as long-term benefits for the environment, saving of natural uranium resources and reduction of the ultimate waste quantities might play an increasing role in a sustainable development context.

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28. Substantial quantities of REPU from MAGNOX fuel have been re-enriched in the UK to natural level for recycling, while the Doel-1 reactor in Belgium has been operated exclusively with re-enriched REPU for a number of years.

### 2.2.2.3.3 CANDU fuel fabrication

Natural uranium fuel is easy to process into fuel assemblies. The short, simple fuel assemblies, called fuel bundles, for the CANDU (PHWR) reactors are easily produced by countries that operate these reactors. Korea, Argentina and Romania all have independent PHWR fuel fabrication facilities sufficient to meet their demands, and China will build a fuel fabrication facility to meet the requirements of their CANDU reactors now under construction. Furthermore, PHWR fuel cycle costs are lower than those of other reactor designs, because natural uranium is relatively inexpensive, the uranium utilisation (amount of energy produced from the mined uranium) is good, and because of the simplicity of the PHWR fuel bundle.

While several fuel bundle designs are in use in PHWRs, a typical fuel bundle design is the CANDU 6, 37-element bundle. Its thirty-seven elements of natural uranium dioxide ( $\text{UO}_2$ ) pellets sheathed in Zircaloy-4 (Zr-4) are held together by Zr-4 end plates. A thin layer of graphite (CANLUB) is applied to the inside surface of the sheathing to reduce the pellet/sheath interaction. The Zr-4 sheathing is closed at both ends by resistance welded end caps that serve three purposes: to provide a seal for the contents of the element, to provide effective element termination for attachment to the end plates, and to provide the structural component for interfacing with the fuel handling system.

The basic simplicity of PHWR fuel has led to extremely low fabrication costs, because only a small industrial factory is required and manpower needs are moderate. The use of natural uranium also eliminates dependence on foreign sources of enrichment (although fuel cycle costs are actually reduced with SEU in the PHWR). Thus, a country with PHWRs has the possibility of becoming independent from foreign nuclear fuel vendors within a fairly short time. In contrast to reactor systems requiring enriched fuel, all manufacturing stages of natural  $\text{UO}_2$  fuel can be accomplished without special criticality restrictions.

### 2.2.2.3.4 FBR fuel fabrication

Liquid metal cooled fast reactor (LMFR) technology has been developed in many countries, including France, Germany, India, Japan, the Russian Federation, the United Kingdom and the USA [28]. The liquid metal mostly used so far is sodium but development is underway in Russia to use lead or lead-bismuth. Other developments (US, France) envisage gas-cooled fast reactors. Most FBR fuel is MOX with Pu content higher than LWR-MOX fuel. Fuel pins of stainless steel cladding tube are assembled in a hexagonal wrapper tube. The attractiveness of FBR-fuel when compared to LWR-fuel is especially related to the effective utilisation of resources, the higher burn-up achievable, higher power ratings and the possibility to burn isotopes that would be neutron poisons to LWR fuels. The fabrication technology is established and fuels were supplied successfully in France, US, UK, Russia and Japan on a pre-industrial scale. In Japan, JNC has fabricated MOX fuel for the experimental FR Joyo and prototype FBR Monju since 1973. The current development of fast reactor technology is rather difficult, particularly in the context of an overall slowing-down of nuclear power development. Today, some increasing interest is going to fast reactors as part of a partitioning and transmutation scheme where new fuel matrices are needed. Next to known fuel fabrication techniques, such as pelletising and vibro-packed fuel, other developments are being tested for industrial deployment in the mid to long term (see Chapter 4).

### 2.2.2.3.5 Thorium fuel fabrication

The fabrication of thorium-based fuel has been performed for years in some countries. The fabrication technology was largely derived from the U- or U/Pu-based fuel fabrication technology. The processes are rather similar in the different countries, because there has been a close co-operation at the research and industrial levels.

Fabrication of thorium subassemblies does not pose any serious radiological problems. However, one of the main drawbacks of thorium-<sup>233</sup>U fuel cycle is the presence of hard  $\gamma$ -emitters (2.5 MeV) among the daughter products of <sup>232</sup>U that is always present with <sup>233</sup>U. This necessitates shielding and remote-handling facilities for manufacture of <sup>233</sup>U-based fuels.

Considerable experience was accumulated in the USA in thorium fuel fabrication. For the Fort St. Vrain high temperature gas reactor, 2 448 hexagonal graphite fuel elements using 26 000 kg of fissile and fertile material in TRISO-coated fuel particles were produced. This included almost 25 000 kg of thorium and performance of the fuel exceeded predictions. It was irradiated at temperatures greater than 1 300°C to a maximum burn-up in the fissile particles of 16% fissions in initial metal atoms (approximately 170 000 MWd/tHM) to a maximum fast neutron fluence of  $4.5 \times 10^{25}$  n/m<sup>2</sup> with no evidence of significant coating failure. Over 50 tons of thorium fuel in ceramic form, clad in Zircaloy, was manufactured for the Shippingport LWBR core. No specific problems were encountered during fuel manufacture. However, US regulations governing maximum permissible concentration (MPC) limits on thorium in air have been dramatically tightened since this fuel was produced and allowable limits have been reduced by a factor of 20. Thus, renewed manufacture of thorium fuel would increase in cost due to tight health physics control and monitoring. In the USA, the Indian Point I ThO<sub>2</sub>/UO<sub>2</sub> core was reprocessed at the West Valley reprocessing plant as a commercial demonstration project. In Germany and the USA, a specific fuel fabrication technology for thorium-based fuels has been developed up to a well proven, industrial scale process under the High Temperature Reactor (HTR) programme. Usually, coated particles were used, i.e. a spherical fuel kernel (oxide or carbide of U, Th or Pu) surrounded by layers of a coating material such as pyrolytic carbon or silicon carbide. Powder agglomeration processes or wet-chemical processes (sol-gel) for the gelation of droplets from a solution containing thorium and uranium could be used to produce the kernels. It was found from the development of prototype plants for the fabrication of mixed oxide fuels (Th/U or Th/Pu) during the 1960s in the USA that the sol-gel process could be a very suitable process although somewhat delicate to use and not suitable for remote manufacturing facilities. A process based on “sol-gel microsphere pelletisation” has also been developed in Germany for the fabrication of high-density oxide pellets for HTRs, at relatively low compaction pressure and low sintering temperature, avoiding dust generation. In India, the use of thorium in reactors has been envisaged mainly in the form of thorium oxide where conventional dry powder metallurgy techniques for compacting and sintering were adopted.

In conclusion, despite possible increases in the costs of manufacturing, the experience gained in Th-fuel manufacturing justifies the pre-industrial status of these processes. Some consideration is given today to use uranium-fuel fabrication plants for Th/U-fuel fabrication in a recycling fuel cycle. One should keep in mind that Th/U-oxide fuels (72 GWd/tHM) show a 50% higher neutron source strength and almost 250% more decay heat compared to standard UOX-fuel of 45 GWd/tHM. However, a massive change to a Th-cycle would demand major investments in all fuel cycle steps, and this is considered to be feasible only in a longer-term perspective.

### ***2.2.3 Handling of the by-products and waste streams***

The back-end of the fuel cycle starts once the fuel has been unloaded from the reactor and resides in the storage-ponds at the reactor site, waiting for spent fuel disposal or reprocessing. Because of the strategic and very long term safety implications related to the spent fuel management, the selection of an option for the back end of the nuclear fuel cycle is an important decision that all countries with nuclear power programmes want to regulate and not leave to the industries alone.

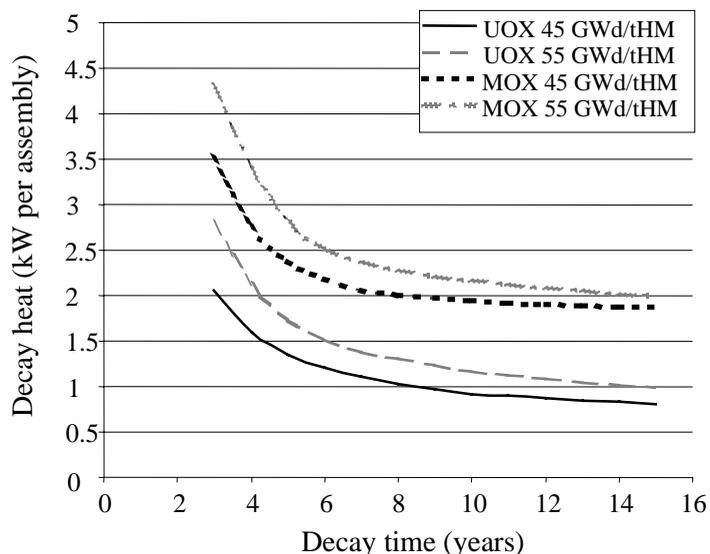
### 2.2.3.1 Spent fuel management

After irradiation in the reactor, spent fuel has to be stored for at least a few years in the spent fuel storage pools at the reactor site in order to allow decay of the short-lived fission products and to reduce the rate of heat production from the spent fuel (Figure 2.5). Thereafter, although the fuel assemblies are still highly radioactive and produce significant quantities of heat, transport of the spent fuel is more readily accomplished. This spent fuel can go either to a reprocessing facility or to away-from reactor long-term storage facilities for an additional period of interim storage.<sup>29</sup>

A number of different approaches, depending on the country's spent fuel management policy, have been developed for long-term interim storage in which the fuel assemblies, either as a whole or after dismantling to reduce their volume, are stored in cooling pools or in dry storage in which the fuel assemblies can be safely held in either an air or an inert gas atmosphere. Today, an increasing use of dry storage, after a short wet storage period, is advocated by the utilities as these techniques have become mature, licensed<sup>30</sup> and are considered to involve low operational costs.

Dry storage casks do introduce some restrictions, dictated by the integrity of the fuel rod cladding tubes, on how and when spent fuel assemblies can be loaded into the cask. Higher burn-up of fuel assemblies leads to higher decay heat levels and higher n- and  $\gamma$ -source terms. This correlation is generally not linear for neutron emitters, since the neutron radiation levels increase disproportionately with fuel burn-up. While the decay heat power levels in the interim dry storage cask affect, in particular, the temperature-dependent behaviour of the fuel rod cladding, the n- and  $\gamma$ -source terms dictate the requirements to be met by the shielding. These two aspects are, in fact, in conflict with each other.

Figure 2.5 **Developments in the front-end (e.g. increasing burn-up or using MOX fuel) have significant consequences for the spent fuel management** [29]



29. The storage of spent fuel at the reprocessing site, in case of a reprocessing policy, is considered as an away-from-reactor wet storage period.

30. Most dry storage casks are licensed for up to 50 years storage.

Typically, the higher dry storage temperature and the increased fission gas release – from the higher decay heat from fuel assemblies with higher burn-up and/or MOX fuel – results in a higher internal gas pressure within a spent fuel rod. This will generate higher stresses and strain in the fuel rod cladding. Since a longer residence time of a fuel assembly in the core tends to decrease the residual wall thickness through in-reactor corrosion, stress and strain is increased even more. In order to stay within the licensed limits of stress and strain, longer residence time of the fuel in wet storage is required or other measures – e.g. specifically optimised storage cask loading patterns – are required for high burn-up fuels and especially for MOX fuels. Figure 2.5 illustrates the consequences of a burn-up increase and/or the use of MOX on the cooling time required to reach a certain decay heat level. It should be noted that, for a 1 300 MWe PWR, each additional year of decay time requires about 50 rack positions in the cooling pool. These developments show the inter-dependency of front and back-end of the fuel cycle, since advances in fuel reliability and burn-up can have an impact on the spent fuel management.

Today's experience indicates that no real technological reasons should limit the use of wet and dry storage for a period of about 100 years. More extended storage periods are being investigated, e.g. in France, in relation to a long deferral of any decision for geological disposal. It should be remarked that comparable and even higher decay heat loads would be encountered in advanced fuel cycles incorporating fast reactor fuel (see Chapter 4).

#### *Once-through fuel cycle option*

In the once-through fuel cycle option, the spent fuel is subsequently transported to a spent fuel conditioning and encapsulation plant before the final disposal in a deep geological facility, designed to retain the fission products and actinides long enough to render any release harmless to the environment. Somewhat different conditioning and encapsulation technologies are being developed in the countries considering the once-through cycle, depending on their final waste repository concept and geology.

In Spain, for instance, the spent fuel would be packed in cylindrical carbon steel disposal canisters, accommodating 4 PWR or 12 BWR spent fuel assemblies (intact) or three vitrified waste canisters, i.e. a total weight of about 15 t. These would be placed in the underground disposal site in granite, clay or salt formation. A decision on the most suitable formation is foreseen after 2010 [30].

In Sweden, the encapsulation plant would take assemblies from interim storage and transfer them to canisters made of copper with a steel insert, change the atmosphere to inert gas, put lids on the canister and seal the lids by electron beam welding. The quality of the filled and sealed canisters would be checked by non-destructive examination methods – ultrasonic and radiographic – before shipping to the repository. Each canister can hold 12 BWR fuel assemblies or 4 PWR assemblies. The copper thickness would be about 5 cm and the steel thickness at minimum about 5 cm. The copper is to be thick enough to prevent corrosion from penetrating the canister as long as spent fuel radio-toxicity substantially exceeds that in a rich uranium ore (i.e. for about 10 000 years). The combined thickness of steel and copper is designed to prevent any significant radiolysis of water outside the canister after deposition in galleries in a granite formation, themselves filled with bentonite clay. The steel insert is intended to withstand the normal mechanical loads that will prevail in the repository, such as hydrostatic pressure and the bentonite swelling pressure. SKB operates an encapsulation laboratory at Oskarshamn to develop the essential technology. The seal welding technology has been demonstrated on a laboratory scale and full-size canisters have been fabricated on a trial scale.

In Canada, the CANDU spent fuel bundles would be sealed in disposal containers, required to be structurally durable for a period of 500 years after emplacement. A container would hold 72 CANDU fuel bundles. In the conceptual design study, a thin-wall packed-particulate container fabricated of 6.35 mm thick Grade 2 titanium was selected. Copper canisters are also being considered. These containers would then be placed in a repository constructed in a granite pluton at a depth of 1 000 m. Finally, a back-fill of clay-based buffer material would seal the disposal gallery.

In Finland, the spent fuel would be encapsulated in copper-steel canisters (wall thickness 5 cm of copper, 5 cm of steel). These canisters would be deposited in vertical holes in a crystalline bedrock repository, several hundred meters deep. The buffer material to be used is highly compacted bentonite, and during the operation phase the disposal tunnels are filled with a mixture of sand and bentonite shortly after the canisters are placed in the holes. This concept is similar to the concept developed in Sweden.

POLLUX repository casks would be used in Germany to encapsulate the spent fuel, after the spent fuel elements would have been disassembled mechanically into fuel rods and structural parts. In order to save space, the fuel rods would be packaged in bins while the structural parts would be packed in baskets. Each repository cask, which is designed to withstand the mechanical stress due to the rock pressure, can accommodate four bins and one basket. The shielded casks are assumed to be placed in drifts in an underground salt-dome repository at a depth of about 870 m. The drifts and emplacement boreholes would be backfilled with crushed salt from the excavation shortly after the placement of the waste packages.

In the USA, the Yucca Mountain site (silicic volcanic rock) in Nevada has been designated by law to be the site where scientific investigations will be conducted to determine whether the site is suitable for the development of a repository. Intact fuel assemblies would be placed in stainless steel canisters with an outer layer of corrosion-resistant Alloy C-22. The canisters can contain up to 21 PWR fuel assemblies. The repository will remain open for a period of decades, with the option to maintain the repository in an open condition for several centuries. This design allows the possibility for active and then passive cooling of the waste packages after emplacement, until they cool to lower temperatures.

The conditioning of spent fuel results in ILW and HLW. In general, 0.2 m<sup>3</sup> of ILW and 1.5 m<sup>3</sup> of HLW (i.e. the conditioned spent fuel) are created per tonne of spent fuel.

### *Reprocessing fuel cycle option*

In order to make use of the residual energy content of the spent fuel, some countries have chosen to reprocess the spent fuel, thereby separating the plutonium and uranium from the fission products and minor actinides. Separation of plutonium from spent fuel could theoretically lead to a decrease in the long-term radio-toxicity per unit power by a factor of about 10. This number originates from the fact that the amount of plutonium present in spent fuel is about 10 times higher than that of the minor actinides, which have a radio-toxicity per unit mass that is approximately equal to that of the plutonium isotopes. This fuel cycle option presents significant sustainable development benefit for countries that are scarce in energy resources. For example, Japan has no indigenous resources of oil, gas and coal and has no overseas pipelines and, therefore, is developing a national strategy that includes MOX utilisation in LWR and FR.

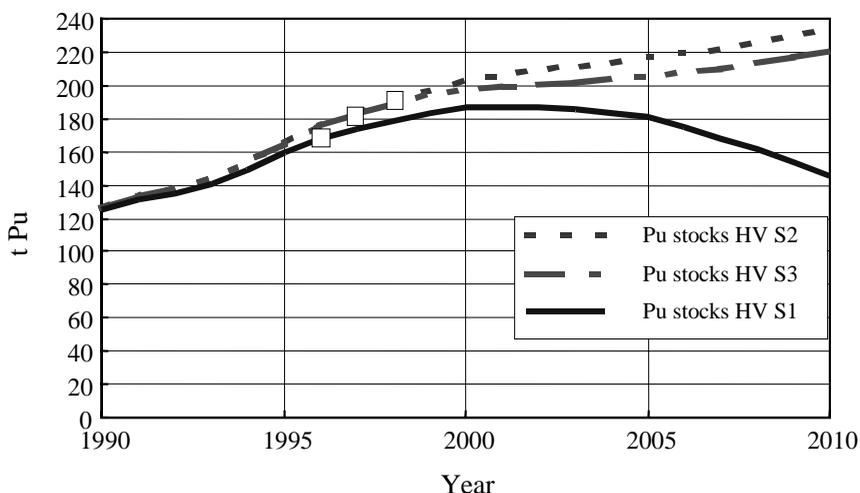
Without fast breeder reactors (FBR), the current method to reuse the separated plutonium is through use of MOX fuel in LWRs. A PWR-MOX fuel assembly of the present design contains 35 kg of plutonium before irradiation and 25 kg after irradiation. This amounts to a net Pu consumption of 10 kg per fuel assembly and per irradiation cycle, compared to a net Pu production of about 5 kg in a standard uranium oxide assembly. At present, a number of reactors in France, Germany, Switzerland and Belgium are being operated with up to 30% of MOX fuel assemblies in the core. However,

recycling of plutonium as LWR-MOX only reduces the radio-toxicity of the assembly by a factor of about 3, if due account is taken of the spent LWR-MOX fuel radio-toxicity, and if it is assumed that this spent fuel is not reprocessed but rather is considered as waste. Current reprocessing practice does reduce considerably the total volume of waste for disposal. Per tonne of spent fuel reprocessed, only 0.115 m<sup>3</sup> of vitrified HLW waste and 0.35 m<sup>3</sup> of ILW waste need to be sent to disposal facilities. In addition, LLW waste undergoes super-compaction, leading to much reduced volume of waste, which then may be included in the ILW waste for disposal.

Forty years of commercial nuclear energy production has resulted in a stock of spent fuel amounting to about 250 000 tHM by the end of 2000. About 10 500 tHM of spent fuel are discharged annually from nuclear power plants, and about 3 000 tHM of spent fuel (about 30% of the total) are reprocessed. Figure 2.6 shows the inventory of separated plutonium resulting from the past and projections into the near future. As shown, those countries that embarked on a Pu recycling policy are facing a stabilisation of their separated plutonium inventory within the coming decade. The imbalance between the separation and use of plutonium as MOX fuel has resulted in an accumulated inventory of separated civil plutonium of about 200 tonnes at the end of 1998.

Commercial reprocessing capacity (see Table 2.7) today amounts to some 3 500 tHM/y. Meanwhile, the BR205 Magnox plant in the UK, with a capacity of 1 500 tHM/y, should be shutdown sometime in the coming decade. World-wide commercial reprocessing capacity after 2015 is forecast to be some 4 300 tHM/y.

Figure 2.6 Separated plutonium inventory<sup>31</sup> [31]



The number of successive Pu cycles in an LWR is limited by the build-up of plutonium isotopes that are not fissionable in the thermal neutron spectrum of LWRs and by the build-up of minor actinides, especially Curium. Increasing the number of MOX assemblies in the reactor core (up to 100%) would lead to a net consumption (rather than a net production) of plutonium during

31. Separated plutonium inventories of the world reported by the Guidelines of Plutonium Management, INFCIRC 549 (open squares) and calculated by the IAEA's VISTA model (solid and broken HV lines). The inputs for calculation are as follows; (1) the world nuclear power capacity grows from 352 GWe in 2000 to 432 GWe in 2010, (2) cases S1 and S2 assume that 50% of spent fuel discharged from NPPs is reprocessed, and case S3 assumes that 35% is reprocessed, (3) S1 assumes that Pu is recycled two times, while S2 and S3 assume that Pu is recycled only once. Taking account of present recycling situation, S3 is the most plausible scenario.

irradiation, but would generally require changes in the moderator/fuel ratio. Higher consumption rate of plutonium can be achieved by mixing the plutonium with a neutron inert material, the so-called inert matrix fuel (IMF), since uranium, the source of the plutonium, is no longer present.

Recycling plutonium from spent LWR-MOX fuel in LWRs seems not to be useful in the next years because of the degraded Pu isotopic composition in the fuel and because reprocessing and MOX fabrication capacities are limited. The LWR-MOX will have to be stored in engineered facilities until disposal. The disposal of such fuel on an industrial scale will have to accommodate an 8.6-fold increase in radio-active inventory with associated heat and criticality limitations. After 30 years, the decay heat amounts to about 2.6 kW/tHM, i.e. still three times as much as for LWR-UO<sub>2</sub> spent fuel. Reduced densities of fuel disposed in the geological site and/or very long cooling times (more than 50 years) before disposal operations start may be needed. Delayed reprocessing of spent LWR-MOX fuel is another option. An option for LWR-MOX reprocessing is the dilution by spent uranium fuel or by reprocessed uranium, which has been demonstrated on an industrial scale by COGEMA and WAK. Particularly, the dilution by spent uranium fuel can improve the degraded plutonium isotopic composition. Another option for reprocessing spent LWR-MOX fuel is to apply FR-fuel reprocessing technology, which can handle fuel of high activity and high Pu content. Multiple recycling of the TRUs after reprocessing to a fast reactor is one of the alternatives before disposal. At this stage in nuclear development, partitioning could play a role in the reduction of the radio-active inventory of HLW glass for disposal.

Table 2.7 Commercial fuel reprocessing plants

| Country                                | Company | Facility/location           | Year of commissioning | Capacity (tHM/yr) | Fuel type  |
|--|---------|-----------------------------|-----------------------|-------------------|------------|
| Plants in operation (01/01/2000)       |         |                             |                       |                   |            |
| France                                 | COGEMA  | UP2-UP3/La Hague            | 1976 and 1989         | 1 700             | LWR        |
| India                                  |         | Prefre-1, Tarapur           | 1974                  | 100               | PHWR       |
|  |         | Prefre-2, Kalpakkam         | 1998                  | 100               | PHWR       |
| Japan                                  | JNC     | Tokai-Mura                  | 1977                  | 90                | LWR, ATR   |
| UK                                     | BNFL    | Thorp/Sellafield            | 1994                  | 1 200             | LWR, AGR   |
|  |         | B205 Magnox                 | 1964                  | 1 500             | Magnox GCR |
| Russian Federation                     | Minatom | RT-1/ Tcheliabinsk-65 Mayak | 1984                  | 400               | VVER       |
| Plants under construction (01/01/2000) |         |                             |                       |                   |            |
| China                                  | CNNC    | Diwopu (Ganzu)              | 2002                  | 25–50             | LWR        |
| Japan                                  | JNFL    | Rokkasho-Mura               | 2005                  | 800               | LWR        |

### 2.2.3.2 Waste disposal

Different waste forms arise from the nuclear fuel cycle. After conditioning operations, three waste types are present:

- Low-level waste (LLW) comprises materials that do not have particularly long decay life nor produce a great deal of heat while decaying.

- Intermediate-level waste (ILW) includes mainly arisings in the fuel cycle facilities, containing medium active waste (Short-lived ILW), although some long-lived  $\alpha$ -emitters may be included (Long-lived ILW).<sup>32</sup>
- High-level waste (HLW) comprises spent fuel or vitrified waste from reprocessing, including the long-lived and higher heat-emitting waste for geological disposal.

Management practices for the disposal of LLW and ILW encompass three main options: near-surface disposal, disposal in caverns at intermediate depth, and disposal in deep geological formations. Near-surface disposal in unlined trenches or pits is an option in a number of countries. This option implies that waste to be disposed contains short-lived radio-nuclides of low or medium specific activity with only very low amounts of long-lived radio-nuclides. Many near-surface repositories in operation or in the design phase are equipped with engineered barriers and drainage systems for control of water infiltration, in order to meet safety regulations or to offer more flexibility for waste packaging. Typically, the disposal units are lined with concrete, bitumen or other materials to improve isolation of the waste. The disposal in caverns at intermediate depth normally implies disposal at a depth of tens of metres, and a variety of cavern configurations located in different geological environments have been used for different types of waste. Typically, these caverns may be classified as specially excavated cavities and disused mines. The sub-surface repositories can in general accept a wide range of radioactive wastes including, for example, long-lived LLW and ILW. Socio-political considerations in certain countries also dictate that LLW and even very-low-level decommissioning waste must also be disposed of in sub-surface caverns. In certain cases, the various waste types are disposed of in separate sections, galleries or vaults. It generally is deemed necessary that HLW and  $\alpha$ -bearing wastes be isolated in deep geological formations, that is at depths of at least a few hundred meters. In certain countries, again on socio-political grounds, deep geological disposal is required for all types of radioactive waste.

The main volume of the waste produced by nuclear electricity production is of LLW-type, which is conditioned before disposal. Concerning ILW and HLW, the nuclear industry has a strong willingness to manage them carefully, and the spent fuels are either stored safely or conditioned and recycled through reprocessing. In addition, the costs for waste-handling and disposal are in most countries covered by the electricity price and no external costs remain. This internalisation of these costs means that enough provisions are available to cover the future expenses for disposal of all types of waste.

Table 2.8 gives an overview of the repositories in operation, under construction or planned in OECD Member countries. Table 2.9 shows the list of underground laboratories for deep geological disposal studies.

The concept of removing long-lived radioactive wastes from the human environment by disposal in deep geological repositories was developed several decades ago. In the intervening years, research efforts world-wide have increased the knowledge and understanding of how underground disposal systems will function over very long periods of time [32]. In recent years, the concept itself has moved closer to implementation, but support is increasingly being voiced for postponement and for more review of alternative waste management solutions. On the other hand, reflections in international groups of experts have repeatedly confirmed the conviction of the waste management community that disposal is ethical, environmentally sound and safe, and that other management options are, at best,

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32. LLW and ILW are defined by the IAEA as radioactive wastes in which the concentration or quantity of radio-nuclides is above clearance levels established by the regulatory body, but with a radio-nuclide content and decay heat level below those of high-level waste (i.e. about 2 kW/m<sup>3</sup>).

complementary to geologic disposal rather than true alternatives. The debates on this subject are not entirely neutral, as the lack of established industrial disposal facilities often is cited by people opposing nuclear development to argue that the waste problem is not solved. Alternatively, this lack is used to justify ever improved conditioning and disposal options, possibly to levels beyond societal significance, with the risk of siphoning R&D efforts off from more fruitful projects for society. Summaries of these reflections are found, for example, in the Collective Opinions published by the OECD/NEA in 1985, 1991 and 1995 [33-35].

In the last decade, significant progress has been made in the technical aspects of geological disposal. In general, the necessary technology for geologic disposal is available and can be deployed when public and political conditions will allow doing so. It is recognised, however, that there is relatively little experience in the application of some of these technologies and therefore demonstration and testing will continue and further refinements will be made. Technical progress has been facilitated through the better integration of the main technical aspects of deep geologic disposal projects, namely the design of engineered systems, the characterisation of potential disposal sites and the evaluation of total system performance.

Important advances have been achieved in the understanding of the performance of system components and their respective roles, the quantitative modelling of their behaviour, treatment of uncertainty, the presentation of assessment findings, and feedback to site selection, characterisation and repository design. Much effort has been focused on increasing the reliability of the methods developed for confidence building/validation. Progress is also evident in the understanding of the natural system and in the characterisation of potential sites. In particular, advances have been made in measurement methodology and procedures, and in the better appreciation of the heterogeneity (spatial variability) of the system.

Over the past decade, some progress has been made towards implementation of repositories. In Germany, a deep repository for disposal of high-level waste and conditioned spent fuel assemblies is being investigated (although exploitation is currently interrupted because of political reasons). The selected geological formation is a salt dome. In Sweden, Finland and Norway, geologic repositories at intermediate depth for low- and intermediate-level waste are now operating in various geological formations, and, in the USA, a deep geologic repository for long-lived waste, the Waste Isolation Pilot Plant WIPP (Carlsbad, New Mexico), commenced operation in March 1999. There have, however, also been delays but there is widespread recognition within the technical community that the critical path towards implementation of disposal facilities is increasingly determined not by technical issues, but by the need for public confidence in the concept.

In Japan, the JNC conducts research and development on underground disposal at the Engineering Scale Test and Research Facility (nicknamed ENTRY) and Quantitative Assessment Radionuclide Migration Experimental Facility (nicknamed QUALITY). In addition, JNC promotes the plan to construct underground laboratories for scientific research on the deep geological environment. JNC, in close collaboration with other organisations, published a progress report on the results obtained in research and development in 1999, in order to evaluate the technological reliability and to provide a scientific and technical basis for the safe disposal of HLW. OECD experts and other researchers made a peer review on this report, and the results confirmed the reliability of deep disposal of high-level waste. In this connection, the Council for promoting high-level radioactive waste Disposal studied what kind of institutional entity should be established, and proceeded with its preparation. Nuclear Waste Management of Japan (NUMO) was established in October 2000 to undertake the construction of the facility for disposing of high level waste and to operate it in about the mid-2030s.

Table 2.8 LLW and ILW repository projects and sites in OECD Member countries [32]

| Country            | Site (starting date–closing date, if applicable)             | Status          | Type of repository |
|--------------------|--|-----------------|--------------------|
| Australia          | Mt. Walton East (1992)                                       | In Operation    | ENSF               |
|                    |  | Site selection  | ENSF               |
| Belgium            |  | Site selection  | ENSF               |
| Canada             | Chalk river  | Under licensing | ENSF               |
| Czech Republic     | Richard II (1964)  | In Operation    | MC                 |
|                    | Bratrstvi (1974)   | In Operation    | MC                 |
|                    | Dukovany (1994)  | In Operation    | ENSF               |
|                    | Hostim (1953-1965)   | Closed          | MC                 |
| Finland            | Loviisa  | Under licensing | MC                 |
|                    | Olkiluoto (1992)   | In Operation    | MC                 |
| France             | Centre de l'Aube (1992)                                      | In Operation    | ENSF               |
|                    | Centre de la Manche (1969-1994)                              | Under closure   | ENSF               |
| Germany            | Konrad   | Under licensing | GR                 |
|                    | Morsleben (1981)   | In operation    | GR                 |
|                    | Asse (1967-1978)   | Under Closure   | GR                 |
| Hungary            |  | Site selection  |                    |
|                    | RHFT Puspokszilagy (1976)                                    | In operation    | ENSF               |
|                    | Solymar (1960-1976)*   | Closed          | ENSF               |
| Japan              | Rokkasho (1992)  | In operation    | ENSF               |
|                    | JAERI, Tokai (1995-1996)                                     | Closed          | SNSF               |
| Korea, Republic of |  | Site selection  |                    |
| Mexico             | Laguna Verde   | Site selected   | ENSF               |
|                    | Maquixco (1972)  | In operation    | SNSF               |
|                    | La Piedrera (1983-1984)                                      | Closed          | ENSF               |
| Norway             | Himdalen   | Under licensing | MC                 |
|                    | Kjeller (1970-1970)**  | Closed          | ENSF               |
| Spain              | El Cabril (1992)   | In operation    | ENSF               |
| Sweden             | SFR (1988)   | In operation    | MC                 |
|                    | Oskarshamn NPP (1986)  | In operation    | SNSF               |
|                    | Studsvik (1988)  | In operation    | SNSF               |
|                    | Forsmark NPP(1988)   | In operation    | SNSF               |
|                    | Ringhals NPP (1993)  | In operation    | SNSF               |
| Switzerland        | Wellenberg   | Site selected   | MC                 |
| Turkey             |  | Site selection  | ENSF               |
| United Kingdom     |  | Site selection  | GR                 |
|                    | Dounreay (1957)  | In operation    | SNSF               |
|                    | Drigg (1959)   | In operation    | E/SNSF             |
| USA                | Other States   | Site selection  |                    |
|                    | Illinois, Ohio, Michigan, New York State, Pennsylvania       | Site selection  | ENSF               |
|                    | Ward Valley, California                                      | Under licensing | ENSF               |
|                    | Boyd County, Nebraska  | Under licensing | ENSF               |
|                    | Wake County, North Carolina                                  | Under licensing | ENSF               |
|                    | Fackin Ranch, Texas  | Under licensing | ENSF               |
|                    | RWMC, INEEL (1952)   | In operation    | S/ENSF             |
|                    | SWSA 6, ORNL (1973)  | In operation    | S/ENSF             |
|                    | Disposal Area G, LANL (1957)                                 | In operation    | SNSF               |
|                    | Barnwell, South Carolina (1971)                              | In operation    | SNSF               |
|                    | 200 East Area Burial Ground, Hanford (1940s)                 | In operation    | SNSF               |
|                    | 200 West Area Burial Ground, Hanford (1996)                  | In operation    | SNSF               |
|                    | Richland, Washington (1965) Savannah River Plant site (1953) | In operation    | SNSF               |
|                    | Environcare  |                 |                    |
|                    | Beatty, Nevada (1962-1992)                                   | Closed          | ENSF               |
|                    | Maxey Flats, Kentucky (1963-1978)                            | Closed          | SNSF               |
|                    | ORNL SWSA 1 (1944-1944)*                                     | Closed          | SNSF               |
|                    | ORNL SWSA 2 (1944- 1946)                                     | Closed          | SNSF               |
|                    | Sheffield, Illinois (1967-1978)                              | Closed          | SNSF               |
|                    | West Valley, New York (1963-1975)                            | Closed          | SNSF               |

\* Waste was moved to another repository (respectively, from Solymar to RHFT Puspokszilagy; and from ORNL SWSA-1 to ORNL SWSA-2).

\*\* Waste will be moved to a new repository (Himdalen) when constructed.

Abbreviations: SNSF = Simple Near Surface Facility; MC = Mined Cavity; ENSF = Engineered Near Surface Facility; GR = Geological Repository; S/ENSF = SNSF and ENSF

Table 2.9 **HLW repository sites and laboratories**

| Country            | Geologic medium                      | Site/Status  |
|--------------------|--------------------------------------|--|
| <b>Belgium</b>     | Clay<br>(plastic Boom clay)          | Mol/Dessel. Specific research on site since 1984.  |
| <b>Finland</b>     | Granite                              | Olkiluoto. Specific research on site since 1992.   |
| <b>France</b>      | Granite                              | Fanay-Augères. Generic R&D. In service from 1980 to 1990.  |
|                    | Clay                                 | Tunnel de Tournemire. Generic R&D. In service since 1992.  |
|                    | Clay                                 | Bure. Generic R&D. Start of construction in 2000.  |
| <b>Germany</b>     | Salt                                 | Asse. Generic R&D. Closed in 1995.   |
|                    | Salt                                 | Gorleben. Specific research on site since 1985. Start of construction of storage in 1997.  |
| <b>Japan</b>       | Granite                              | Mizunami. Generic R&D. Specific research on site since 1996.   |
|                    | Tertiary sedimentary rock (Mudstone) | Horonobe. Generic R&D. Specific research on site since 2001.   |
| <b>Sweden</b>      | Granite                              | Stripa. Generic R&D. In service from 1976 to 1992.   |
|                    | Granite                              | Äspö. Generic R&D. Start of construction in 1990.  |
| <b>Switzerland</b> | Granite                              | Grimsel. Generic R&D. In service since 1983.   |
|                    | Clay (opalinus clay)                 | Mont Terri. Generic R&D. Started in 1995.  |
| <b>USA</b>         | Salt (layers)                        | Carlsbad, New Mexico. WIPP. Specific research on site. In service since 1982 (650 m depth). In exploitation since March 1999 ( $\alpha$ -bearing military TRU-wastes without heat-generation).           |
|                    | Welded tuff                          | Yucca Mountain. Exploratory Studies Facility. Specific research on site (300 m depth). Start of experiments <i>in-situ</i> in 1996. Construction of tunnel for experiments <i>in-situ</i> ended in 1998. |

In many programmes, emphasis is being placed upon the contribution of the engineered barriers, but the natural or geologic barriers in a deep repository plays a crucial role in determining the achievable long-term safety.

Within this context, it should also be noted that waste minimisation programmes have continuously decreased the amount of waste going to storage. This is especially important in the case of LLW and ILW for near-surface or cavern-based disposal, as they involve larger volumes than HLW. The volume of LLW per unit of electricity varies among countries, partly because of different waste treatment approaches and partly because of the different scope of wastes defined as LLW. Whereas Belgium reports a volume of LLW per unit of electricity generated as 10 m<sup>3</sup>/TWh (excluding decommissioning waste), Spain reports 94 m<sup>3</sup>/TWh but including decommissioning waste [36]. A reference value, reported by IAEA [37], of 18.6 m<sup>3</sup>/TWh should be taken as guideline.

The NEA Radioactive Waste Management Committee has identified a few broad strategic areas which should be addressed in the coming years. These areas relate to environmental, safety and sustainable development concerns, where comparison of the principles of radioactive and non-radioactive waste management should be included and where also economic concerns, i.e. the impact of waste management on the continued economic sustainability of nuclear power, should receive increased attention. A second area relates to the process of repository development for long-lived radioactive wastes and also to the management of materials from decommissioning and dismantling and of very low level waste (VLLW). This latter area is important in relation to safe, practicable, cost effective and environmentally sound solutions for VLLW, for which clearance and exemption levels are central issues in the discussion. Improving public perception and confidence is another area for increased focus by the nuclear community. Finally, international guidance and agreements on radiological policy applied to waste management, further system analysis, and technological advances for information exchange within the community are significant activities aiming towards implementation of appropriate waste management policies [38].

In conclusion, the waste arising from the nuclear fuel cycle is well taken care of and technical solutions are either in place or under advanced consideration for implementation. While no specific problems occur for LLW and ILW waste disposal, public concern has delayed the implementation of HLW disposal projects.

### 2.3 Transport

Transport of nuclear materials, whether mined uranium or high-level waste, is a very important activity in the nuclear fuel cycle, subject to stringent national and international regulations (IAEA). It has, however, become the target for socio-political concern because it attracts the attention of many members of the public along the transportation routes. Yet, in contrast to the public perception, nuclear transport is a very safe and mature business, as shown by the experience in many countries. The regime for nuclear transport obeys two fundamental principles:

- The required levels of safety are obtained by use of very well adapted concepts of containers, independent of the safety offered by the transport means themselves.
- The potential risk defines the required level of safety for the container.

In order to evaluate the safety of the transport, consideration is given to risk of collisions, be it due to the transport vehicle or the fall of other objects onto the container, risk of fire, explosion and the combined action of fire and other failure modes, risk of immersion in water, and others. This results in the concept of specific container types for the transport of radioactive materials according, in principle, to two categories, i.e. type A and B. Type A containers accept low level radioactive materials, while type B containers are licensed to accept the higher activity level materials. Next to these two categories, some material of very low level of radioactivity may be transported by normal industrial containers adapted to the transport conditions. These are solid and liquid LLW, uranium minerals, concentrated powder of non-reprocessed natural uranium and depleted uranium.

In France, for example, each year about 15 million packages of dangerous goods are transported, of which some 300 000 packages contain radioactive materials.<sup>33</sup> Of those 300 000 packages, 15 000 relate to the nuclear fuel cycle (the others being radioactive sources for medical, industrial or other use) and only 750 contain fresh or spent fuel or HLW. Statistical data on nuclear transports in France during the period 1975-1997 indicate the occurrence, on average, of only one incident per year with some possible local impact, i.e. contamination of the transport container. Not one incident with radiological consequence has ever occurred in the OECD Member countries during the transport of fresh fuel, spent fuel or HLW. The EC ExternE study (see Section 3.5.3.1) indicated that the transportation of radioactive materials between the different sites, together with the transportation of the materials involved in construction and decommissioning of the reactors, result in a total number of only 0.0003 deaths and 0.0015 injuries per TWhe, all arising from normal accidents involving transport vehicles (i.e. not related to the nuclear nature of the materials transported).

### 2.4 Economics

The 1994 NEA study on *The Economics of the Nuclear Fuel Cycle* [14] showed a small fuel cycle cost advantage of the direct disposal option versus disposal via reprocessing. However, this difference was considered to be insignificant against the background of the underlying cost

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33. In the USA, each year a total of 100 000 transports are performed containing material relating to the nuclear fuel cycle. In Germany, 800 000 packages containing radioactive materials are transported each year, with the overwhelming part relating to medical applications.

uncertainties, country-specific financial, technical, political and environmental boundary conditions and constraints. A sensitivity analysis carried out for the reference cases resulted in overlapping values for the costs of the reprocessing and the direct disposal fuel cycle option. The study concluded therefore that there is no significant difference in the levelised fuel cycle costs of the once through and the reprocessing/recycling option. Several national studies that have been undertaken in the meantime have raised some questions about the perceived costs and benefits of both fuel cycle options. There is no universal answer to this question as the economics are strongly dependent on national (and international) circumstances, policies and boundary conditions. Any fuel cycle cost comparison should take into account the financial aspects, i.e. level of funding and management of provisions, differences in discount rate, and local constraints and issues. Therefore, as international comparisons of levelised fuel costs must, by definition, be based on generic assumptions and some simplifications, the results from such comparisons should only be used as a general indication of trends in the cost of the fuel cycle.

The market situation has changed since the NEA study was published in 1994, and more competition in the front-end and more pressure on cost-reduction in the back-end have combined to reduce fuel cycle costs. In addition, market restructuring has given birth to new companies after mergers and acquisitions.

The following paragraphs will give an overview of the costs of the different fuel cycle steps as reported in literature and by the expert group members as well as possible tendencies for future developments of these cost positions.

The very low natural uranium prices prevailing at the end of year 2000, i.e. about 20 USD /kgU<sub>3</sub>O<sub>8</sub>, imposed economic difficulties even for the very best mines. While a continued supply of “demilitarised” uranium inventories is likely to maintain this low price in the short term, a rebound of demand for newly mined natural uranium (due to draw-down of inventories and other market factors) could result in price increases. A minimum price range of 40-50 USD /kgU<sub>3</sub>O<sub>8</sub> likely would be needed to allow the mining industry to expand production in order to cope with a renewed rapid growth of nuclear power capacity.

The conversion market, in essence based on chemical processes, has experienced a period of decreasing prices in the past years. The present price for long-term contracts for conversion of natural uranium oxide to uranium hexafluoride for enrichment lie in a nominal range of 4-6 USD /kgU, while spot prices in 2000 (reported by NUKEM) lie in the range of 2.45-3.85 USD /kgU. In the longer term, an upper bound of about 8 USD /kgU is estimated.

As indicated above, the enrichment market has seen significant changes and is characterised by persistent over-capacity. This has resulted in a present price range of about 80-100 USD /SWU. In the years to come, the market situation will continue to be influenced by highly-enriched uranium coming from disarmament programmes, which will induce continuous downward pressure on enrichment prices.

Existing excess capacities in a highly competitive market have led to a drastic decrease in the UOX fabrication price, currently in a range between 200 and 300 USD/kgU. With respect to the future development of the UOX fabrication price, the most important factors are technical developments influencing the fuel assembly demand (e.g. burn-up increase), continued efforts to further improve the efficiency of the manufacturing processes, as well as effects resulting from mergers of suppliers (e.g. reduction of excess capacities). In contrast to all other steps in the fuel cycle, fuel assembly design and fabrication also influences the specific costs of the other steps and, being the link between fuel cycle and nuclear power plant, may influence the remaining elements of the power generating costs as well.

Concerning reprocessing, the situation is different, as the market is characterised by a situation with only two main commercial actors and relying on long term contracts with certain utilities. New contracts, making use of existing facilities, seem to indicate significant price reductions, taking benefit from the accumulated experience and reflecting that much of the investment costs have been amortised already. In the future, the new plants would benefit greatly from the large experience that has been gained during the last decades, allowing to simplify the plants, decrease their size, reduce maintenance requirements, etc. If, however the separation of selected nuclides (e.g. minor actinides) would be implemented, the cost could be increased relative to conventional U/Pu separation.

The conditioning and geological disposal of high-level waste (vitrified or spent fuel) does not yet rely on industrial experience and most costs quoted are based on estimates and detailed design studies in the different Member countries. Disposal of HLW is claimed often, by the countries that have nuclear power programmes, as being too important to be left to the producers of the wastes alone and is considered a national responsibility, with the waste producers to pay for the proper disposal of the waste. The handling and disposal of this waste is paid by provisions established by the utilities, while national waste management agencies have been established to perform the disposal operation. As the geological conditions and amounts of waste differ according to the national nuclear energy programmes, the cost ranges are fairly wide. Important technical factors that affect costs are the size of the system, time schedule of the disposal project, geological medium and the barrier system chosen. Next to these technical factors, social and political issues also impact on the costs. These will affect the siting and licensing process as well as the overall waste management policy [40]. Studies show the variability of normalised costs depending on the size of the system and the waste management policy. Recent studies in Belgium have indicated that a very important decrease in cost has been achieved over the past years. Disposal costs are estimated to be some 0.2 MUSD/m<sup>3</sup> or less for spent UOX fuel, and about 0.5-0.7 MUSD/m<sup>3</sup> for HLW [9, 41]. It is important also to consider that the volume of HLW conditioned in glass is about ten times lower than the equivalent spent fuel in a metallic canister. Thus, expressed as cost per kWh of electricity produced, disposal as vitrified HLW is cheaper than disposal as spent fuel. Regarding spent MOX fuel, the cost depends mainly on the decay heat level as a higher decay heat level demands significantly wider spacing of the waste containers. In the case of rapid disposal (i.e. after a short decay time) the heat level could be three times higher, and the cost higher by a similar ratio, than for disposal after a significant period of storage and decay.

In order to give the reader a general framework on the costs of different aspects of the nuclear fuel cycle, Table 2.10 gives an overview of the lower and upper bounds of unit costs expected to be applicable in the short to medium term. Over the long term, beyond 20 years, cost evolution of industrial stages are likely to follow the historic trends, i.e. decreasing in constant terms. The magnitude will depend, however, on aspects related to the vitality of the nuclear industry. Also, fuel cycle cost comparisons need to take into account the financial aspects, such as level of funding, management of provisions and differences in discount rate. Depending on the special boundary conditions, fuel cycle costs for individual countries, and, in addition, for the individual utilities within a country, may deviate significantly from such generic figures.

In conclusion, this chapter has presented information on the current status and short-term trends in the nuclear fuel cycle. As noted in the introduction to the chapter, the currently existing reactors and fuel cycle operations were developed in response to specific needs, and in the context of the criteria that applied at the time they were being developed. However, there now is a need to apply different criteria, in particular taking into account the high importance currently given to sustainable development, conservation of resources, protection of the human and natural environment, etc. Therefore, the report now turns to a discussion, in Chapter 3, of some criteria and analysis methods that can be used for the evaluation of nuclear energy as a sustainable energy source. Application of such criteria and methods may lead to strong interest in advanced reactors and fuel cycle options, such as those that are presented in Chapter 4.

Table 2.10 Expected lower and upper bounds to unit costs for specific fuel cycle activities\*

| Parameter            | Unit                                | Lower bound | Upper bound | Description   |
|----------------------|-------------------------------------|-------------|-------------|---|
| $Cost_U$             | USD/kgU <sub>3</sub> O <sub>8</sub> | 20          | 80          | Unit cost of natural uranium  |
| $Cost_{Uconv}$       | USD/kgU                             | 3           | 8           | Unit cost of conversion   |
| $Cost_{Uenr}$        | USD/SWU                             | 80          | 120         | Unit cost of enrichment   |
| $Cost_{UOXfab}$      | USD/kgUOX                           | 200         | 300         | Unit cost of UOX fuel fabrication   |
| $Cost_{MOXfab}$      | USD/kgMOX                           | 1 000       | 1 500       | Unit cost of MOX fuel fabrication   |
| $Cost_{UOXrepro}$    | USD/kgHM                            | 500         | 900         | Unit cost of UOX fuel reprocessing (as long as percentage of MOX-fuel is less than 20-30 %) |
| $Cost_{MOXrepro}$    | USD/kgMOX                           | 500         | 900         | Unit cost of MOX fuel reprocessing  |
| $Cost_{UOXIntstore}$ | USD/kgUOX                           | 100         | 300         | Unit cost of UOX fuel interim storage (2 years) <sup>34</sup>                               |
| $Cost_{UOXgeo}$      | USD/kgUOX                           | 300         | 600         | Unit cost of UOX fuel geological disposal   |
| $Cost_{HLWgeo}$      | USD/kgUOX                           | 80          | 200         | Unit cost of HLW geological disposal  |
| $Cost_{FR-MOXfab}$   | USD/kgMOX                           | 1 200       | 2 000       | Unit cost of FR-MOX fuel fabrication (including fertile blankets)                           |
| $Cost_{FR-MOXrepro}$ | USD/kgMOX                           | 1 000       | 2 000       | Unit cost of FR-MOX reprocessing (+200 to 300 USD/kgFR-MOX for blankets)                    |

\* Note that some values were converted from Euro to USD, using a conversion of 1 Euro = 1 USD.

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34. The costs for interim storage of spent UOX-fuel were also reported in [112-114], ranging from 40 to 80 USD/kgU, where an interim storage time of two years is standard. Reference [113] also reports a cost for “away-from-reactor” wet storage of LWR fuel assemblies (in 1987-USD) to be a fixed 50 USD/kgHM plus 5×T USD/kgHM within a range of plus or minus 50 per cent, where T(years) is the period of storage. Spent-fuel transport costs have been reported in many publications and amount about 50 USD/kgHM (40 to 60 USD/kgHM).

### **3. CRITERIA FOR NUCLEAR ENERGY AS A SUSTAINABLE ENERGY SOURCE**

#### **3.1 Introduction**

The issue of suitable criteria and indicators for the evaluation, from a sustainable development perspective, of technological systems or sub-systems, in this case for the nuclear fuel cycle, is a subject of intense discussions in academia. The topic is of utmost importance, as often such criteria are inadequately selected and used to draw conclusions that quite likely would have been different if more appropriate criteria and indicators would have been used.

While the Expert Group cannot claim to be fully representative of the multiple interested parties involved in decision making or evaluation of technological systems, in this report on nuclear energy and energy options, the Group did make a concerted effort, by taking an approach that goes beyond only the economic or technical aspects, to lay a basis for a wider decision-aiding technique, i.e. multi-criteria analysis (MCA), that could be used in such decision making. Representatives of the various strata of the society (stakeholders) could use the MCA technique for assessing the sustainability characteristics of a particular project or system. The search for common indicators to compare the different impacts of human activities, e.g. nuclear versus coal, is complicated by the variety of economic, social and environmental impacts to be considered. The impacts might be different in geographical extent, in duration and in phenomenology, and thus their measurement and valuation might be very difficult; nonetheless, individuals, firms and governments do take decisions, and this means that they carry out some kind of valuation of the impacts, however implicit or intuitive the valuation process might be. The goal of explicit valuation is to make the factors going into decision making more transparent and more quantitative. This chapter tries to identify criteria and indicators that are deemed by the Expert Group to be most relevant to nuclear energy in a sustainability context. The other chapters will try to quantify some of these indicators.

#### **3.2 Development and use of criteria and indicators**

An adequate set of evaluation criteria and indicators must itself fulfil a series of requirements that have certain characteristics:

- Indicators should be measurable and quantifiable (i.e. allowing interpretation) and logically independent (to avoid “double counting”). The measurability requirement implies that a certain hierarchy “decomposition” of the criteria is necessary in order to be able to distinguish between various effects and to quantify them.
- The number of indicators must be manageable. Thus, a limit should be put on the aforementioned decomposition, which should not become too detailed. This criterion is driven primarily by the necessity of simplifications in the interactions with stakeholders; from an analytical point of view, it would be possible to handle a much higher level of complexity.

- Despite the limitation above, the selected criteria should be inclusive, i.e. no criteria should be excluded a priori only because in the eyes of individual evaluators they have low relevance. The stakeholders may have a totally different opinion, and they would always have the option to drop certain criteria from the list or to disregard them by assigning a zero weight to them.
- Putting the criteria into operation calls not only for defining quantifiable indicators, but also for pragmatic definitions. Simplifications are necessary with a view to the availability of data, transparency of the process, and the nature of the specific application for which the criteria list is used. If necessary, complexity can be increased at a later stage by expanding the set of criteria, or defining more representative indicators, or both.
- It is not feasible to define a unique set of criteria that would be valid for all applications.<sup>35</sup> Evaluating various electricity supply options based on different fuels is a much different problem than trying to decide which alternative configurations of the nuclear fuel cycle are more economic, or more sustainable. In the latter case, a more detailed distinction between the various nuclear-specific features becomes necessary. The time horizon selected for the evaluation has a similar impact on the set of criteria, i.e. whether systems of a certain “generation” are compared only among themselves, or compared also with systems of other “generations”.
- The structure of the list of criteria may be dependent on the goals of the assessment, the particular issues that should be emphasised, the tools used and the way foreseen for interactions with stakeholders (including the eliciting of preferences). It should be noted that the responses on preferences are not independent of the criteria and hierarchy used in defining the options. Thus, the structure used for communication purposes might not be the same as that used in the assessment process.
- Special difficulties associated with non-quantifiable criteria (e.g. those on social impact) must be recognised and spelled out. In some cases, the scoring is subjective and might not be within the competence field of the analysts involved. The inclusion of such criteria and sub-criteria is highly dependent also on the goals and expected output of the evaluation. One could claim that, for individuals, such criteria are binary (black/white; go/no-go), and that political processes leading to decisions take full care of such criteria. One possible solution consists of including them only in a subsequent phase of the evaluation, e.g. in a sensitivity check.

### ***3.2.1 Sustainable development criteria***

For the purpose of this study, and based on the comprehensive definition of sustainable development – considering economic, environmental and social aspects, sustainability criteria have been grouped into three categories or principles:

- “No” degradation of resources in the broadest sense.
- “No” production of “non-degradable” waste.
- High potential for robustness/long-term stability.

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35. OECD developed a set of sustainable development indicators according to five categories (see <http://www.oecd.org/env/>): Summary indicators; Outcome indicators; Resource indicators; Sectoral indicators; Sub-national (community) indicators.

In the above list, “no/non” means that the characteristic should be as small or as near to zero as possible. Each principle is related to a set of specific criteria and indicators, which aim at being representative rather than complete. Table 3.1 summarises a set of representative criteria selected to assess energy-related technologies against sustainability. The context is taken to be a large regional and global scale, and the time frame assumed is the mid-term, i.e. the years 2020-2030 and beyond.

An assessment based on such criteria should be normalised, e.g. quantity per TWhe electrical energy output, taking into account the whole energy chain including fuel front- and back-end and other elements of the life cycle (Life Cycle Analysis approach).

**Table 3.1 Set of criteria for evaluating energy-related technologies against sustainability**

|   |
|---|
| <p><i>“No” degradation of resources</i></p> <ul style="list-style-type: none"> <li>– Fuel: resources depletion time.</li> <li>– Land: used to support “normal operation”, contaminated areas due to “severe accidents”.</li> <li>– Environment: emissions (GWP-CO<sub>2</sub> equivalent, acid rain-SO<sub>x</sub>).</li> <li>– Human health: irreversible damage after ‘severe accidents’; risk estimates (normal operation and accidents).</li> <li>– Social aspects: public acceptance, proliferation, economics (competitiveness, investment needs).<sup>36</sup></li> <li>– Employment effects.</li> </ul> |
| <p><i>“No” production of “non-degradable” waste</i></p> <ul style="list-style-type: none"> <li>– Amount and characteristics of radioactive waste.</li> <li>– Required secure confinement times.</li> </ul>  |
| <p><i>High potential for robustness/long-term-stability</i></p> <ul style="list-style-type: none"> <li>– Security of supply: influence of geopolitical regions; assured technology availability.</li> <li>– Failure friendliness: “grace periods”, need for safety culture, sensitivity to socio-political changes.</li> </ul>  |

A suitably filled-in matrix, accepted by consensus among stakeholders, could be a tool to assess and cross compare current technologies, as well as more advanced or even potential options, in the context of sustainable development. The weighting and aggregation of the proposed indicators and their subsequent use within decision-making processes need to be developed based on a case-by-case application.

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36. Some experts consider public acceptance, for example, also as a resource [40].

### 3.2.2 Proposed list of criteria

Based on the general set of criteria described above with regard to technologies fitting into sustainable development and following the basic principles governing a sound set of criteria and indicators, a list of criteria for the assessment of nuclear fuel cycles in the present study was developed. Some other criteria (e.g. precautionary principle, retrievability, licensing and regulatory process) were discussed, but finally were considered to be developments or changes in the (policy) boundary conditions rather than additional criteria. Changes in the required characteristics of geological disposal (retrievability), and in the regulation and licensing of disposal facilities, are trends in policy and practices, and thus are not distinctive criteria. “Safety” could be similarly addressed as well, taking into consideration that nuclear power has to be safe, and that the ensuring of safety induces costs and is translated into time requirements for R&D, licensing, etc.

The discussion on criteria, related indicators and corresponding aggregation/evaluation approaches in this study aims:

- At demonstrating that it is possible to base future decisions on a relatively small set of consistent criteria which are quantifiable and manageable, and, thus, to show a possible practical approach for use by decision makers.
- To indicate that the quantification of proposed indicators both for current and future systems aims at providing a traceable “validation” of the conclusions presented in Chapter 6, especially regarding potential developments, the exploitation of which would largely improve the degree to which nuclear fuel cycles satisfy sustainability requirements.

On the other hand, it was not the intent to use the quantified indicators to suggest that a particular concept or technology is better or worse than another one and to recommend particular technological policies. The scope of the study, and also the resources available, were too restricted to enable a serious and comprehensive life cycle analysis (LCA) or to permit an aggregation of the individual indicators for all future concepts discussed herein. It was intended rather to demonstrate that such an approach is possible – if desired – with currently available assessment and decision aiding tools.

The corresponding indicators for this study were divided into four types:

- Type A identifies key indicators, which generally are aggregated. The description and assessment of technological developments and trends in the fuel cycle have been, thus, considered especially from the perspective of these indicators.
- Type B denotes quantifiable indicators that correspond to main technological features of the fuel cycle. Such indicators are partial components of type A indicators and allow to better understand their nature.
- Type C refers to qualitative indicators that are based on quantitative data but also on rather empirical judgement; in particular, their interpretation is based not only on quantitative constituents but also on personal preferences.
- Type D comprises indicators that are in principle quantifiable, but for which there are no significant differences between the fuel cycle options examined in this study; they were, therefore, not included in the detailed analysis performed.

The criteria finally taken into account in this study are summarised in Table 3.2.

Table 3.2 List of selected indicators

|  | Criterion                           | Type                  | Indicator   | Measure  |
|--|-------------------------------------|-----------------------|---|--|
| <i>Economy</i>                         | Economic competitiveness            | <b>A</b>              | Levelised fuel cycle cost   | USD/kWh (100%)   |
|  |                                     | <b>B</b>              | Raw materials(U <sub>3</sub> O <sub>8</sub> or Th)<br>Separation work<br>Conversion<br>Fabrication<br>Storage<br>Reprocessing<br>Transports<br>Encapsulation and conditioning<br>Disposal | USD/t (%)<br>USD/SWU (%)<br>USD/kgHM (%)<br>USD/kgHM (%)<br>USD/kgHM (%)<br>USD/kgHM (%)<br>USD/kgHM (%)<br>USD/kgHM (%)<br>USD/kgHM (%) |
|  | Financial expenditure               | <b>A</b>              | Total cost  | USD  |
|  | Technology availability             | <b>A</b>              | Research (govt.)<br>Development (non-govt.)<br><b>I:</b> basic R&D<br><b>II:</b> laboratory/process<br><b>III:</b> pre-industrial<br><b>IV:</b> industrial                                | USD<br>USD<br>years  |
| <i>Environment and Public Health</i>   | Use of non-renewable resources      | <b>A</b>              | Energy recovered per kg U   | TWh/kg U   |
|  | Energy intensity                    |                       | Ratio of necessary energy input to obtained output  | %  |
|  | Transportation                      | <b>D</b>              | Range of ton-kilometres<br>Energy intensity ratio   | t×km/kWh<br>%  |
|  | Land occupation                     | <b>D</b>              | Land area used  | km <sup>2</sup> /kWh   |
|  | Greenhouse gas emission             | <b>D</b>              | Amount of GHG emitted   | t CO <sub>2</sub> eq./kWh  |
|  | Amount of waste                     | <b>A</b>              | Total volume<br>Volume at each life cycle step  | m <sup>3</sup> /kWh<br>m <sup>3</sup> /kWh   |
|  |                                     |                       | α-emitters<br>γ-emitters  | kg/kWh; Bq/kWh<br>kg/kWh; Bq/kWh   |
|  | Confinement time of waste           | <b>A</b>              | α-emitters<br>γ-emitters  | years<br>years   |
|  |                                     |                       | Radiological impacts  | <b>C</b>   |
| Human health effects: acute fatalities | <b>D</b>                            | Operational accidents | # of immediate fatalities per accident  |  |
| <i>Society</i>                         | Human resources, work opportunities | <b>A</b>              | Change in work opportunity<br>Work opportunity  | person×year/kWh<br>person×year/kWh   |
|  |                                     |                       | Broad economic effects  | <b>D</b>   |
|  | Social aspects                      | <b>A or C</b>         |   |  |
|  | Proliferation resistance            | <b>A</b>              |   |  |

### 3.2.3 Description of selected criteria

A brief description of some of the criteria listed in the Table 3.2 is given below, along with some remarks on their application.

- **Economic competitiveness** is expressed, if possible, in terms of full levelised fuel cycle cost. It should be based on generic cost data, such as *The Economics of the Nuclear Fuel Cycle* [14]. Instead of absolute cost data, which could be difficult to gather and subject to commercial confidentiality restrictions, an attempt should be made to express the costs in relative terms, i.e. as sensitivity with regard to different parameters. The different steps of each fuel cycle are the components determining its costs; they are mentioned explicitly and measured according to the acknowledged commercial units.
- **Financial expenditure** describes the investments needed (excluding R&D) to develop the necessary infrastructure for a specific fuel cycle, i.e. relating to the degree of capital intensity for development. This value is of interest, in case one cannot gather specific cost data for each step of a particular fuel cycle. However, such a criterion remains fuzzy as long as boundary conditions are not accurately defined. Most of the other criteria discussed here are values that may ultimately be expressed in terms of a unit of energy produced. On the contrary, financial expenditure should reflect whether capital requirements could become so high as to inhibit the introduction of a new technology at commercial scale. This criterion is independent of the amount of fissile material treated in the corresponding facilities and of the amount of energy subsequently produced therefrom. A possible way out of this dilemma would be to calculate the capital requirement for a first-of-a-kind facility with a minimum capacity still to be agreed upon.
- **Technology availability** can be expressed (a) in terms of necessary R&D investments until the specific technology is fully available on an industrial scale; a further refinement can be achieved by differentiating between investments for Research (Governmental funding) and for Development (usually carried out by the private sector once the technology has achieved an industrial stage); (b) in terms of time elapsed before industrial employment. A combined assessment can be used based on a qualification of a specific fuel cycle in four groups (from I to IV), where I stands for basic R&D, II for laboratory/process development, III for pre-industrial development and IV for available and industrialised technology. It is usual in the nuclear field for an application, that is technologically available, to require an additional five to ten years, before it can be commercially implemented, because any change in a nuclear application has to be demonstrated to be safe in the view of the regulatory authorities, and achieving regulatory approvals can take that order of time period. For applications that are at earlier stages of development, completion of each stage typically may imply a latency time between ten and twenty years.
- **Use of non-renewable resources** is expressed often in terms of depletion time for each raw material involved in the specific fuel cycle. However, this approach would imply making assumptions on levels of existing resources. As this study deals with one basic material, uranium, it is preferable to express the economic usage of this resource in terms of energy recovered per kilogram of natural uranium.
- **Energy intensity** describes the energy input required over the whole life cycle per unit of energy produced and is expressed as a percent ratio of the two. Energy associated with transportation throughout the life cycle of a specific fuel chain can be used here as a representative example for the associated energy intensity.
- **Land occupation** corresponds to the soil surface that, during the lifetime of a specific installation, is not available for other uses. As this surface does not vary significantly for different nuclear fuel

cycles, this criterion is not used in this study. However, land occupation is a typical example of a controversial indicator. In ecologically driven models, land occupation is defined as a product of the area occupied and the time for which the land is removed from other uses. In this context, there is a differentiation between various states of the land (e.g. natural, modified, industrialised), depending on the degree of human intervention. Transfers between these states are addressed through consideration of specific time durations, depending, in turn, on the nature of the human intervention. Nuclear industry uses very small land areas which, if not restored to the original natural state, are barred from other uses for extremely long times determined (like production of many other toxic elements) by the long decay times of the waste products that have low levels of radioactivity. However, these areas usually are in rather isolated locations, and there is, therefore, little impact on humans.

- **Confinement time of waste** is determined by the decay time of the specific radio-nuclides contained in the waste and expresses the time lag necessary for the waste to reach a state in which its influence on the biosphere cannot be differentiated from the effects of natural radiation. To allow a better characterisation of each fuel cycle, the confinement time should be given separately for  $\alpha$ -emitters and  $\gamma$ -emitters and for each fuel cycle step. The radiological burden not to be exceeded in the biosphere is determined by protection goals defined at national or international (e.g. ICRP) level.
- **Radiological impacts** are regarded here only from the point of view of operational exposure, expressed in terms of collective dose.<sup>37</sup> As certain fuel cycles are coupled to a specific reactor technology (e.g. liquid metal cooled fast reactors, gas cooled reactors etc.), the operational exposure resulting from the use of a specific fuel in the corresponding reactor concept must be explicitly considered for this criterion. Operational exposures (either to nuclear workers or to the general population) can be varied to a very large extent by the regulations imposed at national levels. As such, this indicator is not truly representative of one fuel cycle compared to another, as ultimately, through technological adaptations (which will eventually show in economic terms), they will have to be made comparable. Public exposure was omitted as a criterion, because under normal operation radioactive releases are limited by international and national regulations, which ensure that no harm will occur to the population. The public would be affected only in the case of accidental releases, which, however, do not differ between the fuel cycles considered in this study.
- **Human health effects** are expressed in terms of acute fatalities resulting from operational accidents. Again, as certain fuel cycles depend on specific reactor technologies, operational accidents coupled to a particular reactor concept must be explicitly considered for this criterion.
- **Human resources, work opportunities:** The impact of a given fuel cycle technology on the work market is expressed in terms of working places in absolute terms and of changes in work opportunities (new working places per year) taking the whole life cycle into account.
- **Broad economic effects** can be taken into account in terms of autonomy of resources (e.g. time of guaranteed power supply by a given quantity of uranium resources) and of induced industrial

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37. The concept of “collective dose” has been questioned recently and there has been advice that objective operational guidance for the valid application of “collective dose” is needed [42]. There have been views that, even for comparative purposes, collective dose should be truncated at some predetermined lower level, and/or after a limited number of years. In addition, disaggregation should be applied, i.e. meaning presentation in terms of individual dose (e.g. average dose to the critical group, maximum dose to the critical group, individual dose distribution curve within the exposed population, etc.), the number of people exposed, the temporal and geographic population, etc.

production (e.g. changes in GNP in the nuclear industry sector that can be attributed to the particular fuel cycle). While this indicator may be rather invariant among different nuclear fuel cycles, the local public and socio-political situation may be very sensitive to this indicator. While this indicator will not be used in this study, broad economic effects are very important from a public acceptance perspective.

- **Social aspects** are addressed in terms of public acceptance/risk aversion. Both of these indicators are essential, especially with regard to the nuclear fuel cycle, where the issues of reprocessing plants, transport of spent fuel or HLW after reprocessing and final repositories for spent fuel or HLW are in the focus of public opposition. However, public acceptance varies significantly between countries and is not always in line with political priorities. Addressing these issues here would be only qualitative and hardly compatible with the technical content of the study. However, the aforementioned aspects, that are likely to influence public acceptance either positively or negatively, are explicitly dealt with for each fuel cycle:
  - Risk aversion against nuclear (and other hazardous technologies) is usually measured by the extent of land losses and/or of fatalities (acute and latent) following a severe accident. The probability of a severe accident in the nuclear fuel cycle outside the energy generation stage (i.e. the power plant) is – although not excluded, e.g. Tokaimura – nevertheless very low and quite independent from a specific fuel cycle.
  - On the other hand, there are issues other than severe accidents that differ considerably from one fuel cycle to another and may influence public acceptance. Three main issues that may belong to this context have been identified:
    1. Necessity for new nuclear facilities in the front-end or back-end of the nuclear fuel cycle.
    2. Number of spent fuel and waste transports (per unit of energy generated).
    3. Volume and confinement time for the final waste.
  - These sub-criteria can be evaluated qualitatively in terms of “very high”, “high”, “medium”, “low” and “no” (i.e. five rating levels) and accordingly each can be rated from +2 to –2. High total scores indicate a high public aversion and, therefore, a low public acceptance.
- **Proliferation resistance:** Proliferation risks are judged differently by different experts, and there also are differences between experts and the public and media. Some see proliferation risks as one of the key issues, while others (in particular, some experts) make reference to the fact that there has been “no-case-of-experience” over the last decades and reject proliferation concerns as a misperceived artificial problem. Indeed, the results of the INFCE study [43] indicate that it is “not possible to make the nuclear fuel cycle absolutely “proliferation-proof” by technical fixes only”, but notes that “integration of reprocessing in the reactor plant is advantageous” and “increased build-up of higher isotopes through multiple recycling makes plutonium less attractive for theft”. Indicators of proliferation resistance could be based on physical characteristics (neutron emission rate, decay heat, ...), but no internationally agreed set of indicators has been developed until now.

Among the above criteria, some can be directly perceived by the public. On the other hand, some are so complex that the broad public cannot understand them fully, even if the mass media put them in the foreground.

- Criteria that are directly perceived as being essential by the broad public are economic competitiveness, greenhouse gas emissions, confinement time of waste, radiological impacts, social aspects and proliferation resistance.

- Criteria that can become perceivable, after sufficient information is provided, are technology availability, use of non-renewable resources, land occupation, amount of waste, human health effects and human resources/work opportunities.
- Criteria not readily understandable by the general public include broad economic effects, energy intensity and financial expenditure.

This categorisation still contains some obvious elements that would influence the relative weights that stakeholders would attribute to individual criteria in a decision aiding process like a multi-criteria analysis. People tend to consider issues that they easily understand as being the most important. However, other factors may influence the relative weights placed on particular criteria. Among such factors are different “values” depending on national backgrounds and priorities but also on the “zeitgeist”.

Different countries tend to have different attitudes towards issues that are driven by economic, social, political or national security priorities (e.g. fears of nuclear weapon proliferation, security of energy supplies), and public attitudes are influenced by the manner in which the issues are presented by the media. In addition, there are attitudes driven by cultural aspects (history, religion, traditions) that may influence the choices of a given society (e.g. Puritanism, which often results in an extreme respect for nature). Finally, temporary conditions may have an impact on the way a social group defines its priorities (e.g. an attitude of “nothing to lose” in a poor society may create a more relaxed attitude towards technical risks). While the first factors are quite long-lasting, the last one may be subject to quite rapid changes.

A similar characterisation and categorisation of the criteria can be made from the engineer’s or scientist’s point of view. From this perspective, attention is paid to those criteria that differentiate one cycle from another. A first indication has been given above by declaring some of the criteria to belong to the type D. However, even among the criteria of type A, there are some that may be decisive in a prioritisation process. Such criteria include economic competitiveness, technology availability, use of non-renewable resources, amount and confinement time of waste and proliferation resistance.

### **3.3 Use of the criteria**

The above description of a set of criteria and indicators introduces questions about the quantification of these indicators and also on their further use. While the Expert Group discussed the set of relevant criteria from a broad perspective, i.e. sustainable development, the group also recognised that full development of these criteria and indicators, i.e. quantification and finally use in a concrete selection and prioritisation process, would be out of the scope of their study. Several reasons may be identified in that respect:

- Firstly, quantification of the criteria and indicators needs a lot of data gathering for use in life cycle analysis, and cross-checking of the data should be a continuous action. Only a few research groups are working currently on this, although a first attempt to gather these data on an international level has been undertaken, i.e. DECADES [44].
- Secondly, almost all of the available data relate to existing industrialised processes and facilities. No cross-checked and validated data are available, for obvious reasons, for more prospective technologies.
- Thirdly, as this study considers only the nuclear sector, there are few of the criteria and indicators that will differ between different options. In addition, the definition of criteria and indicators for other energy sectors (oil, gas, coal, renewables) has not been reported yet and,

therefore, placing these rather indifferent criteria only within a nuclear perspective would not be productive for a better understanding of nuclear vis-à-vis other options.

While recognising the need to consider all criteria and indicators in a study addressing energy policy in a sustainable development context, the Expert Group considered that their study only could highlight, and where possible quantify, the main trends and developments in the fuel cycle that would respond to the concerns of the different stakeholders, as expressed by the set of criteria and indicators chosen.

This study is related to two techniques or instruments, i.e. multi-criteria analysis (MCA) and life cycle analysis, without however originally applying these techniques. The study has essentially used the results of life cycle analysis work performed by other groups, and has not engaged in performing a new life cycle analysis of its own. The same applies for the multi-criteria analysis, as the expert group limited itself to a definition and analysis of the applicability of criteria (indicators) and the description of the different fuel cycle options and developments according to these criteria. In doing so, the expert group aimed to provide a basis for a multi-criteria analysis without pretending to aggregate the indicators in a decision making process. For reasons of completeness, and also to indicate possibilities for the continuation of this process, the following paragraphs will briefly describe these two techniques.

### ***3.3.1. Life cycle analysis***

The kind of assessment called life cycle analysis consists of using a universal set of criteria to quantify and compare the multiple impacts that energy systems exert on their surroundings, through all stages of their existence (i.e. from cradle to grave or “life cycle”) [45, 46]. These issues were first identified during the energy debate in the 1970s, where they were summarised by Sørensen [47]. In depth studies were carried out during the 1980s by OECD in its COMPASS project [48], and by UNEP [49]. Important ingredients in these studies were the many projects on risk analysis conducted in the 1970s and early 1980s.

A potential weakness of LCA is the tremendous amount of data involved. Indeed, it is very difficult to document clearly and understandably all data and assumptions that go into the final results from any life cycle analysis. In the (rather likely) case that different studies for the same products would come up with different final results, one should be able to trace these discrepancies back to the different assumptions made. This is possible only if all base data are accessible and well documented as regards their sources. In its application, LCA is an inventory method and, as a result, generates a long list of substances that are:

- Produced by the system studied either as useful products or wastes discharged into the environment.
- Consumed by the system studied either in material or energy form.

In order to use LCA results for policy decisions, the lists of substances need to be interpreted and, in general, reduced to a limited number of factors. This reduction can be done by using one or several of the substances as indicators of the overall impacts or by aggregating the list of substances into limited numbers (e.g. total material requirements, total air emissions, ...). Both methods involve some value judgement. Questions arise subsequently about the comparability of the various substances and their relative harm to the environment.

One of the most extensive and very well documented life cycle analysis studies is the *Ökoinventar von Energiesystemen* performed by ETH-Zurich, PSI and Bundesamt für Energie (Bern) in Switzerland [50].

### *Approach to accounting for technological progress in the LCA of energy systems*

Particular attention is needed when dealing with the performance of concepts, systems or cycles to be implemented in the future. The LCA methodology was developed and applied primarily for the analysis of existing technologies and technological systems. Consequently, the input data normally are based on experience accumulated with these systems, and the standard approach is static. Therefore, applications to future systems require extensions, extrapolations and a number of additional assumptions. The extensions of LCA applications to future systems is based on literature, direct information from the industry and consultants, and expert judgement.

For example, of particular importance in the study reported in the *Ökoinventar* study [50] was the input received from ABB on coal and gas power plant technologies, from BNFL and COGEMA on some steps in the nuclear chain, and from a solar company on photo-voltaic. Availability of essential, LCA-specific process information and knowledge about the relative importance of the various sources of emissions made it possible to focus the analysis and economise the use of resources. The main driving parameters in the LCA study are: emissions, efficiencies, material intensities (for construction and operation), and transportation requirements. The relative importance of these parameters varies significantly between energy chains.

For nuclear technology, the most important expected improvements of ecological performance identified and evaluated were: reductions of long-term radon emissions from mine/mill tailings, reductions of electricity consumption in enrichment by replacement of diffusion by centrifuges or laser technologies, power plant improvements (particularly extended life time and increased burn-up), use of modern reprocessing facilities, and reduced volume of conditioned radioactive solid wastes.

For electricity inputs needed for the LCA-modules external to Switzerland, an average European mix for the year 2010 was used, based on a forecast by the OECD International Energy Agency. As compared to the current situation the mix reflects an expansion of gas, reduction of oil shares and a relatively small but significantly increased contribution of photo-voltaic. Coal, hydro and nuclear remain at about the same level. Since the “new” systems generally show better performance and lower emissions than the “old” ones, assumptions needed to be made with respect to the market penetration of the “new” ones. This was done individually for each energy source taking into account their specificity and the expected developments reflected by the assumed overall contribution of each energy carrier to the mix.

The future cost analysis also builds on literature studies and inputs from manufacturers. In addition, for systems currently having small market shares but large development potential, learning curves are used to account for improved economic performance, given projected major increases in production volumes.

### **3.3.2 Multi-criteria analysis**

In order to cope better with the inherent value judgements when interpreting LCA data, the use of decision-aiding methods is appropriate. Multi-criteria analysis (MCA) is an appropriate decision-aiding technique which can supplement LCA techniques.

The balancing of different criteria in a decision making process demands not only the input of validated data (e.g. from LCA) but also a methodology to assess the various perceptions of different stakeholders involved in the decision making process. This is needed especially when consideration should be taken of the context of the decision framework.

Classic modelling approaches, where one would try to model rather complex situations, sometimes requiring the use of overly simplistic assumptions, are not suitable for dealing with multi-criteria socio-political problems. Therefore, the use of rather simple calculations, while retaining as much as possible the complexity of the real problem, seems to be a more appropriate approach. Generally speaking, in a multi-criteria approach the objective is to find a satisfactory solution to the problem in question, that is, a solution that is “most adequate” or “as favourable as possible”, considering the whole set of criteria which have to be taken into account [51].

The application of multi-criteria analysis demands two conditions: firstly, multiple criteria or indicators should be available and assessed; secondly, multiple stakeholders should be present in order to evaluate the weighting factors of the different criteria. A good introduction to the MCA methodology, its potential and the problems associated its application can be found in [52-55,115].

### **3.4 Quantification of criteria for current nuclear fuel cycles**

Table 3.3 gives an overview of available quantitative data, selected from the literature, for the different criteria and fuel cycle steps. These data have been collected mainly from the Ökoinventare project, and have been updated according to new studies in the field.

One should, however, keep in mind that these data are indicative only, and sometimes are based on different hypotheses and models, and new operational practices in the fuel cycle could make some of the data values obsolete. In addition, as table 3.3 indicates, further work is necessary, preferably on an international level, to update and harmonise the data, in order to have validated and agreed data that are the most appropriate for multi-criteria analysis in the energy field.

### **3.5 Overview of related studies**

Several studies on fuel cycle issues have been published in the past years. Most of these studies were addressing a specific dimension (economic, environmental, safety, ...) of the fuel cycle, and only a few addressed a wider perspective. This section will give an overview of the main studies published in the past few years.

#### ***3.5.1 Sustainable development and related indicators***

In 1998, the OECD launched a three-year “horizontal project” on sustainable development. The project is intended to provide an integrated framework to address policy issues of interest to governments of OECD countries, including their interactions with industry and with non-member countries. The project aims toward making the sustainable development concepts operational for public policies and should help Member countries to address fundamental sustainable development issues.

The OECD Environment Directorate proposed a commonly agreed set of environmental indicators (see Tables 3.4 and 3.5) for OECD countries and for international use. It covers issues that

reflect the major environmental concerns in OECD countries. Workshops and conferences are organised on the measurement of sustainable development, and the discussions showed considerable interest in indicators that better reflect the linkages between the three dimensions (economic, environment, development) of sustainable development and the interactions (positive and negative) of these dimensions. Interest has been expressed in reduced and balanced sets of indicators that provide signals on key dimensions of sustainable development to policy makers and the general public. Core sets of indicators provide a base of comparable information to which countries can add country-specific indicators to suit their circumstances. Core sets can be adapted also for different purposes (e.g. tracking performance against plans; providing budgetary information). The OECD conferences and workshops also highlighted, but did not reach conclusions on, four key areas [56]:

- First, the importance of a pragmatic approach to developing indicators (i.e. ensuring that they are useful to decision makers and to stakeholders).
- Second, it was emphasised that work on communicative indicators needs to be linked more closely to an understanding of citizens' interests and information needs.
- Third, it was suggested that there is a need to demonstrate the “value-added” that indicators have for advancing sustainable development, especially in order to secure resources for further work, at the national level and by international organisations.
- Finally, work is required to strengthen links between institutions, in particular those with expertise on social issues, in light of the need for indicators that reflect better the social dimensions of sustainable development.

### ***3.5.2 General studies on the nuclear fuel cycle***

The International Nuclear Fuel Cycle Evaluation (INFCE) [43] was conducted during the period 1978-1980, and covered a technical and analytical study of the nuclear fuel cycle. Sixty-six countries and five international organisations contributed to the study, which was co-ordinated by the IAEA. The scope of the study was to take a forward look into the development of nuclear energy, and the different options that could be implemented in the context of a perceived massive deployment of nuclear energy in order to meet the world's energy requirements. One of the main guiding principles of the study related to the proliferation resistance of the different technology options and fuel cycle strategies, when implemented in such a large-scale deployment of nuclear energy. However, the conclusions of the study indicated that effective political measures can and should be put into place at national level and through international agreements to minimise the danger of proliferation of nuclear weapons without jeopardising energy supplies or the development of nuclear energy for peaceful purposes. This conclusion indicated that proliferation resistance as such should not be considered as a key indicator for fuel cycle options. The measures to secure non-proliferation, on the other hand, could be seen as a key indicator, as more or less stringent measures might be demanded for different fuel cycles, and these could have an impact on the technological and economic viability of developments in the fuel cycle.

Environmental, health and safety issues were addressed by INFCE to determine whether specific fuel cycle activities could be carried out in conformity with accepted standards, and the study reported this to be the case. There was, however, no use of life cycle analysis or other techniques to describe the sustainability impacts and, in general, the descriptions were mainly qualitative.

Table 3.3 Overview of quantitative criteria for fuel cycle steps

|  |                                     |  |  |                          | Uranium Mining                          |  | Milling                                    |
|--|-------------------------------------|--|--|--------------------------|---|--|--|
|  |                                     |  |  |                          | Open Pit                                | Underground                                |  |
|  | Criterion                           | Type                                   | Indicator  | Measure                  |   |  |  |
| Economy                                | Economic competitiveness            | A                                      | Levelised fuel cycle cost                          | USD/kWh                  | -                                       |  | -  |
|  |                                     | B                                      | Unit cost  | USD/kgU                  | 20-80                                   |  |  |
|  | Financial expenditure               | A                                      | Total cost   | USD                      |   |  |  |
|  | Technology availability             | A                                      | Maturity   |                          | IV                                      | IV   | IV   |
|  |                                     |  | R&D spending                                       | USD/year                 | 80 10 <sup>6</sup> -130 10 <sup>6</sup> |  |  |
|  | Use of non-renewable resources      | A                                      | Energy recovered per kg                            |                          | -                                       |  |  |
|  |                                     |  | Materials use                                      |                          |   |  |  |
|  |                                     |  | Concrete   | kg/kgU                   | 9.1 10 <sup>-4</sup>                    | 9.6 10 <sup>-1</sup>                       | 2.1  |
|  |                                     |  | Steel  | kg/kgU                   | 2 10 <sup>-1</sup>                      | 6 10 <sup>-1</sup>                         | 2.2 10 <sup>-1</sup>                       |
|  |                                     |  | Noble metals                                       | kg/kgU                   | 4.1 10 <sup>-3</sup>                    | 4.9 10 <sup>-3</sup>                       | 6.4 10 <sup>-3</sup>                       |
| Others                                 |                                     |  |  |                          |   | 4.4 10 <sup>-3</sup>                       |  |
| Water                                  | kg/kgU                              | 3 10 <sup>3</sup> -9.3 10 <sup>3</sup> | 8.1-35   | 700-1 800                |   |  |  |
| Environmental and Public Health        | Energy intensity                    | A                                      | Ratio of necessary energy input to obtained output |                          |   |  |  |
|  |                                     |  | Electricity  | kWh/kgU                  | 4.3 10 <sup>-6</sup> -1.3               | 3.8 10 <sup>-6</sup> -1.5                  | 3.4 10 <sup>-5</sup> -7.9 10 <sup>-5</sup> |
|  |                                     |  |  | TJ <sub>e</sub> /kgU     |   |  |  |
|  |                                     | Fossil fuel                            | TJ/kgU   | 1 10 <sup>-5</sup>       | 1 10 <sup>-4</sup>                      | 3.4 10 <sup>-4</sup> -5.7 10 <sup>-4</sup> |  |
|  | Transportation                      | D                                      |  | t-km/kgU                 | 8.6                                     | 1.6  | 0.11-8                                     |
|  | Land occupation                     | D                                      |  | m <sup>2</sup> -year/kgU | 1.7-22.5                                | 6.7 10 <sup>-2</sup> -0.15                 | 8.2 10 <sup>-2</sup> -1.5 10 <sup>4</sup>  |
|  | Greenhouse gas emission             | D                                      | SO <sub>2</sub>                                    |                          |   |  |  |
|  |                                     |  | NO <sub>x</sub>                                    |                          |   |  |  |
|  |                                     |  | CO <sub>2</sub> eq.                                |                          |   |  |  |
|  | Amount of waste                     | A                                      | Total  | kg/kgU                   | 21 10 <sup>3</sup>                      | 1.5 10 <sup>3</sup>                        |  |
| Air emission of U-α emitters           |                                     |  |  |                          |   |  |  |
| Water emission of U-α emitters         |                                     |  |  |                          |   |  |  |
| LLW                                    |                                     |  |  |                          |   |  |  |
| Confinement time of waste              | A                                   |  |  |                          |   |  |  |
| Collective operational dose            | C                                   |  | manSv/kgU  | 5.53 10 <sup>-5</sup>    |   |  |  |
| Human health effects: acute fatalities | D                                   |  |  | 1.19 10 <sup>-7</sup>    |   |  |  |
| Society                                | Human resources, work opportunities | A                                      |  | man-year                 | 54 000                                  |  |  |
|  | Broad economic effects              | D                                      |  |                          |   |  |  |
|  | Social aspects                      | A                                      |  |                          |   |  |  |
|  | Proliferation resistance            | C                                      |  |                          |   |  |  |

Table 3.3 Overview of quantitative criteria for fuel cycle steps (contd.)

| Conversion             | Enrichment                                 |  |        | Fuel Fabrication       |             |
|------------------------|--|--|--------|------------------------|-------------|
|                        | Diffusion                                  | Centrifuge                                 | Laser  | UOX                    | MOX         |
| –                      | –  |  |        |                        |             |
| 3-8                    | 80-120                                     |  |        | 200-300                | 1 000-1 500 |
| IV                     | IV   | IV   | III    | IV                     | IV          |
| –                      |  |  |        |                        |             |
| 4.5 10 <sup>-2</sup>   | 2.1  | 3.4  |        | 1.6                    |             |
| 1.9 10 <sup>-2</sup>   | 0.68                                       | 0.91                                       |        | 0.43                   |             |
| 7.3 10 <sup>-4</sup>   | 6.2 10 <sup>-2</sup>                       | 1.7 10 <sup>-2</sup>                       |        | 1                      |             |
| 3.2                    | 5.7 10 <sup>-3</sup>                       | 6.4 10 <sup>-2</sup>                       |        | 4 10 <sup>-2</sup>     |             |
| 5.0 10 <sup>2</sup>    | 3 600                                      | 110  |        | 300                    |             |
|                        |  |  |        |                        |             |
|                        | 2 900                                      | 260  | 50-120 | 8 10 <sup>-5</sup>     |             |
| 3.7 10 <sup>-5</sup>   | 1 10 <sup>-2</sup>                         | 9.4 10 <sup>-4</sup>                       |        | 1 10 <sup>-4</sup>     |             |
| 7 10 <sup>-4</sup>     | 3.6 10 <sup>-4</sup>                       | 2.1 10 <sup>-4</sup>                       |        |                        |             |
|                        | 1.46                                       | 2.13                                       |        | 2.9                    |             |
|                        |  |  |        | 0.24-0.33              |             |
|                        | 7.6 10 <sup>-2</sup> -40                   | 7.6 10 <sup>-2</sup> -9.6 10 <sup>-2</sup> |        |                        |             |
|                        | 6 10 <sup>-2</sup> -11                     | 6 10 <sup>-2</sup> -7.6 10 <sup>-2</sup>   |        | 3.0 10 <sup>-2</sup>   |             |
|                        | 1.2 10 <sup>-3</sup> -2.7 10 <sup>-1</sup> | 1.2 10 <sup>-3</sup> -2.1 10 <sup>-3</sup> |        |                        |             |
|                        |  | 5.2 10 <sup>-1</sup>                       |        |                        |             |
|                        | 1.8 10 <sup>-1</sup> -6.9 10 <sup>-1</sup> | 1.8 10 <sup>-1</sup> -2.7 10 <sup>-1</sup> |        | 0.13                   |             |
|                        | 1.1 10 <sup>-2</sup>                       | 1.9 10 <sup>-2</sup> -2.9 10 <sup>-2</sup> |        | 7.13                   |             |
| 3.0 10 <sup>-2</sup>   | 4.7 10 <sup>-4</sup>                       | 1.3 10 <sup>-3</sup>                       |        | 2.7 10 <sup>-4</sup>   |             |
|                        |  |  |        |                        |             |
| 8.54 10 <sup>-10</sup> | 4.32 10 <sup>-9</sup>                      |  |        | 2.14 10 <sup>-9</sup>  |             |
| 4.27 10 <sup>-11</sup> | 2.16 10 <sup>-10</sup>                     |  |        | 1.07 10 <sup>-10</sup> |             |

Table 3.3 Overview of quantitative criteria for fuel cycle steps (contd.)

|  |                                     |        |  |                                  | Reprocessing   | Interim                     | Storage                        |
|--|-------------------------------------|--------|--|----------------------------------|--|-----------------------------|--------------------------------|
|  |                                     |        |  |                                  |  | ILW                         | HLW                            |
|  | Criterion                           | Type   | Indicator  | Measure                          |  |                             |                                |
| Economy                                | Economic competitiveness            | A      | Levelised fuel cycle cost                          | USD/kWh                          | 500-900  | IV                          | IV                             |
|  |                                     | B      | Unit cost  | USD/kgU                          |  |                             |                                |
|  | Financial expenditure               | A      | Total cost   | USD                              | IV   | IV                          | IV                             |
|  | Technology availability             | A      | Maturity   |                                  |  |                             |                                |
|  | Use of non-renewable resources      | A      | Energy recovered per kg                            |                                  | 8.3-35.9<br>0.72-3.79<br>$9.7 \cdot 10^{-3}$ - $1 \cdot 10^{-2}$<br>650-1 500          | 710<br>44 kg/m <sup>3</sup> | 3 500<br>220 kg/m <sup>3</sup> |
|  |                                     |        | Materials use                                      | kg/m <sup>3</sup>                |  |                             |                                |
|  |                                     |        | Concrete   | kg/kgU                           |  |                             |                                |
|  |                                     |        | Steel  | kg/kgU                           |  |                             |                                |
|  |                                     |        | Noble metals                                       | kg/kgU                           |  |                             |                                |
|  |                                     |        | Others (b)   | kg/m <sup>3</sup>                |  |                             |                                |
| Water                                  | kg/kgU                              |        |  |                                  |  |                             |                                |
| Environmental and Public Health        | Energy intensity                    | A      | Ratio of necessary energy input to obtained output |                                  | $2.6 \cdot 10^{-4}$ - $9.4 \cdot 10^{-4}$<br>$1.2 \cdot 10^{-3}$ - $2.7 \cdot 10^{-3}$ | $3.2 \cdot 10^{-2}$         |                                |
|  |                                     |        | Electricity  | TJ <sub>el</sub> /m <sup>3</sup> |  |                             |                                |
|  |                                     |        |  | TJ <sub>el</sub> /kgU            |  |                             |                                |
|  | Fossil fuel                         | TJ/kgU |  |                                  |  |                             |                                |
|  | Transportation                      | D      |  | t-km/kgU                         | 76<br>34   | 186<br>170                  |                                |
|  | Land occupation                     | D      |  | m <sup>2</sup> -year/kgU         |  |                             |                                |
|  | Greenhouse gas emission             | D      | SO <sub>2</sub>                                    | kg/kgHM                          |  |                             |                                |
|  |                                     |        | NO <sub>x</sub>                                    |                                  |  |                             |                                |
|  |                                     |        | CO <sub>2</sub> eq.                                |                                  |  |                             |                                |
|  | Amount of waste                     | A      | Total  | t/kgU                            | 165  |                             |                                |
|  |                                     |        | Air emission of U-α emitters                       |                                  | $3.2 \cdot 10^2$   |                             |                                |
|  |                                     |        | Water emission of U-α emitters                     |                                  |  |                             |                                |
|  |                                     |        | LLW  |                                  |  |                             |                                |
| Confinement time of waste              | A                                   |        |  |                                  |  |                             |                                |
| Collective operational dose            | C                                   |        | manSv/kgU  | $2.4 \cdot 10^{-3}$              |  |                             |                                |
| Human health effects: acute fatalities | D                                   |        |  | $1.2 \cdot 10^{-4}$              |  |                             |                                |
| Society                                | Human resources, work opportunities | A      |  | man-year                         |  |                             |                                |
|  | Broad economic effects              | D      |  |                                  |  |                             |                                |
|  | Social aspects                      | A      |  |                                  |  |                             |                                |
|  | Proliferation resistance            | C      |  |                                  |  |                             |                                |

Table 3.3 Overview of quantitative criteria for fuel cycle steps (contd.)

| Conditioning  | Disposal   |     |  |                   |
|---|--|-----|--|-------------------|
|   | LLW  | ILW | HLW  | Spent Fuel        |
|   |  |     | 80-200 USD/m <sup>3</sup>  | 300-600 USD/kgUOX |
| IV  | 1 000-7 000 USD/m <sup>3</sup> -capacity                               |     | III-IV   |                   |
|   | 5 400<br>120 kg/m <sup>3</sup>   |     | 5 600<br>5 900 kg/m <sup>3</sup>   |                   |
|   | 1 200<br>450   |     | 62 000<br>5 200  |                   |
| 2.5   |  |     |  |                   |
| 3.6 TJ/m <sup>3</sup><br>210 t-km/m <sup>3</sup>                            | 1 749 t-km/m <sup>3</sup><br>15-25 m <sup>2</sup> -year/m <sup>3</sup> |     | 8 433 t-km/m <sup>3</sup><br>360-380 m <sup>2</sup> -year/m <sup>3</sup> |                   |
| 5.04 kg/m <sup>3</sup><br>21.6 kg/m <sup>3</sup><br>1 138 kg/m <sup>3</sup> |  |     |  |                   |
| 602 TBq/m <sup>3</sup>  |  |     |  |                   |
|   | 2.86 10 <sup>-4</sup> manSv/m <sup>3</sup>                             |     | 2.72 10 <sup>-1</sup> manSv/m <sup>3</sup>                               |                   |
|   | 1.42 10 <sup>-5</sup>  |     | 1.36 10 <sup>-2</sup>  |                   |

Table 3.4 Core list of OECD environmental performance indicators

|                  |                       |   |
|------------------|-----------------------|---|
| Pollution issues | Climate change        | CO <sub>2</sub> emission intensities      |
|                  | Ozone layer depletion | Ozone depleting substances                |
|                  | Air quality           | Air emission intensities                  |
|                  | Waste                 | Waste generation intensities              |
|                  | Water quality         | Waste water treatment collection rate     |
| Resource issues  | Water resources       | Intensity of use of water resources       |
|                  | Forest resources      | Intensity of use of forest resources      |
|                  | Biodiversity          | Protected areas                           |
|                  | Land resources        | Changes in land use and in key ecosystems |
|                  | Energy resources      | Intensity of use of energy resources      |
|                  | Mineral resources     | Intensity of use of mineral resources     |

Table 3.5 Core list of OECD energy-environment indicators

|   |  |  |
|---|--|--|
| Sectoral trends of environmental significance | Overall energy use   | Total primary energy supply, consumption per fuel type, consumption per sector   |
|   | Energy intensities   | Total primary energy intensity per unit of GDP, sectoral end uses, fossil fuel efficiency for electricity generation   |
|   | Energy mix   | Total primary energy supply by source  |
|   | Energy use   | Total final energy consumption by sector   |
| Interactions with the environment             | Proven oil/coal/gas/... reserves in tonnes of oil equivalent |  |
|   | Air pollution  | Energy related air emissions and intensities   |
|   | Water pollution  | Oil released through accidents, on a continuous basis  |
|   | Waste  | Volume of solid waste, radioactive waste (spent fuel) from power generation  |
|   | Land use   | Hectares of land taken up by energy production, transport, and transformation  |
|   | Safety   | Numbers killed or injured  |
| Economic and policy aspects                   | Environmental pollution damages                              | Relating to energy production and consumption, for certain types of pollutants (e.g. SO <sub>x</sub> )   |
|   | Expenditure  | Expenditure on pollution prevention and clean-up, abatement vs. clean technology, public vs. private.  |
|   | Pricing  | Real energy prices per fuel type   |
|   | Subsidies and taxation                                       | Direct subsidies per fuel type, total economic subsidies (direct and indirect subsidies plus externalities), relative taxation in percent by different fuel types, subsidies to energy sector and energy users |

### 3.5.3 *Life cycle analysis of nuclear energy*

#### 3.5.3.1 *ExternE*

At the beginning of the 1990s the debate about the environmental and economic effects of energy use was turning into a question of internalising the externalities (i.e. those costs or impacts not borne directly by the producers and consumers of energy, but rather by society as a whole), as a means to assist policy and decision making. In order to provide answers to some of the questions raised in this debate, the European Commission and the US Department of Energy launched a joint research project to assess the environmental externalities of energy production and use.

A common methodology, based on scientific and economic information, was set-up and has been developed continuously by the European Commission since 1994. Today, a rather comprehensive set of data is available, and a rather large number of researchers from different disciplines continue to collaborate in order to integrate existing, but dispersed, research results into a coherent framework. The different fuel cycles (nuclear, coal and lignite, oil and gas, wind and hydro) were analysed in the study, and the results were reported in a series of reports [39]. An update of the methodology and data was published in 2000.

The nuclear part of the ExternE study was based on the nuclear fuel cycle of France, which was broken down into 8 separate stages for purposes of the study [39, Vol. 5]. Reference sites and technologies in use in the 1990s were chosen to represent the total nuclear fuel cycle. In addition, the transportation of material between the sites was considered. The facilities were assessed for routine operation, except for the power plant and transportation steps, where accident situations were evaluated also.

The main objective of the nuclear studies within the ExternE project was to develop an impact pathway methodology for the nuclear fuel cycle that would be consistent with the methodologies developed for other fuel cycles. Within the context of the ExternE project, it was not possible to consider all of the impact pathways, and therefore, only the most important impacts, called priority impacts, were included. Releases of radioactive material to the environment, which potentially impact public health, were given the highest priority. Occupational health impacts, from both radiological and non-radiological causes, were the next priority.

This ExternE project delivered a rather detailed assessment of the radiological impact of the nuclear fuel cycle, accounted in economic values. It did not, however, address other dimensions of the nuclear fuel cycle, such as non-radioactive wastes and areas for potential improvements, although information on these dimensions could be needed in a decision-aiding methodology for analysing the role of nuclear energy in sustainable energy development.

#### 3.5.3.2 *Ökoinventare*

An impressive work has been undertaken in Switzerland in the field of life cycle analysis covering the nuclear fuel cycle as well as other energy systems (oil, natural gas, coal, hydro, wood, geothermal, solar thermal, photo-voltaic). This study was first published in 1994, with average systems existing in the year 1990 taken as reference; a first update was published in 1996 [50] referring to systems of the early 90s; a second update to be published by end of 2002 is underway currently, referring to energy systems of the late 90s. The calculation of the energy chains is done from the perspective of both the average Swiss and European final use.

The main scope of the study was to provide environmental information on existing energy systems. In order to achieve a high degree of transparency and objectivity, the data are presented in a disaggregated format. These data, describing energy inputs to industrial and residential sub-systems, can be used for the LCA of products and for other environmental assessment studies. One limitation of the study, for purposes of comparative assessment of energy systems, is that the data are limited to current technologies. With regard to the nuclear fuel cycle, the study includes all fuel cycle steps for both PWR and BWR, including reprocessing, with detailed LCA data from different sources. As “rare” accidents with an occurrence  $<10^{-3}$  are not included in the study, nuclear accidents and large oil spills have not been considered.

The LCA data used in the present report have been taken from the “Ökoinventare” study, and additional input from other LCA studies or updates were added to represent “current best practice”.

### **3.5.4 Technical/economic studies**

#### *3.5.4.1 Economics of the nuclear fuel cycle*

The open and mono-recycling closed fuel cycle (as well as the CANDU fuel cycle) were analysed from an economic perspective in the 1994 NEA study *The Economics of the Nuclear Fuel Cycle* [14]. Based on a methodology of levelised lifetime fuel cycle cost calculation, this study indicated no significant economic difference between the open and closed fuel cycle options. In addition, it showed that a 40% reduction in projected fuel cycle costs (in constant money values) had occurred during the period 1984-1994.

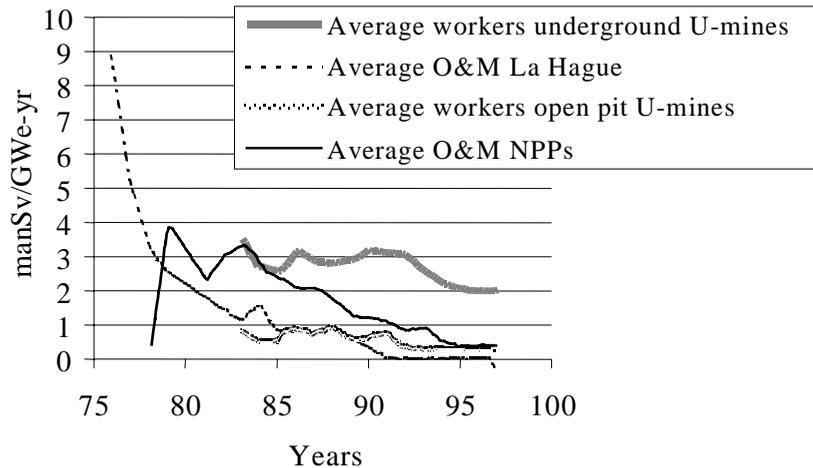
Since this study focused on the economic dimension only, it did not include any description of longer term developments in the fuel cycle nor of other environmental or social considerations. The conclusion of the study suggested that considerations of national energy strategy – including reactor type, environmental impact, balance of payments and public acceptability – would play a more important role in deciding a fuel cycle policy than would the small economic difference reported.

#### *3.5.4.2 Safety of the nuclear fuel cycle*

The safety aspects of the nuclear fuel cycle were addressed by NEA in a study published in 1993 [57]. A detailed assessment was made of the different steps in the fuel cycle and an overview of critical safety aspects was presented. The general conclusion of this study indicated that considerable development had occurred since the early 1980s (previous evaluation of fuel cycle safety by the NEA). The study also indicated the occurrence of a shift in emphasis, from technical aspects of the nuclear facility safety towards the radiological impacts of those facilities on the personnel and environment.

In general, the report also indicated that adequate protective measures were taken for criticality control, avoidance of diversion of fissile material, technical accidents and the impacts on the environment, and that this was illustrated by the continuous compliance with regulatory limits. Figure 3.1 indicates, for example, the reduction of collective doses to workers at the La Hague reprocessing plant, in comparison to workers in uranium mines and nuclear power stations [58].

Figure 3.1 **Annual doses to workers at the La Hague reprocessing plant in comparison to doses to uranium miners and staff of NPPs**



#### 3.5.4.3 Broad economic impacts of nuclear power

The 1992 NEA report on *Broad Economic Impact of Nuclear Power* [59] complemented other work on the projected costs of electricity generation, by defining and listing factors, related to nuclear power, which are taken into account by governments and utilities but which are not explicitly considered in electricity production costing. This report attempted to illustrate the macroeconomic factors in the decision-making process of various countries. Macroeconomic impacts discussed were: employment, balance of payment, security of supply, as well as environmental, health and socio-cultural issues.

The conclusions of this report indicated that the NEA's comparative cost analysis for future generation plants (such as the recent update [12]) includes all significant incremental financial costs for the different options considered. For modern nuclear plants, the potential health and environmental costs were reported as small (<1 %) relative to the direct generation costs, even when account was taken of hypothetical large-scale low-probability accidents. The same does not apply for fossil-fired plants where the environmental and health costs are not fully reflected in the generation costs (especially in relation to greenhouse gas emissions).

The study attempted to put the macroeconomic costs in perspective and to describe and comment on the methods that have been applied for their evaluation. These costs were categorised according to three categories:

- Externalities: where the actions of the electricity generators impose costs or benefits on others with no compensating payments, e.g. acid rain or climate change impacts.
- Strategic factors: e.g. security of energy supplies, long-term adequacy of fossil-fuel supplies.
- Macroeconomic factors: e.g. benefits to employment, price stability, economic growth.

#### 3.5.4.4 Health and environmental aspects of nuclear fuel cycle facilities

The IAEA undertook a study during 1992 to 1994 on the health and environmental aspects of the nuclear fuel cycle. The report [60] served as a kind of introduction on the subject and was related to another initiative undertaken by several international organisations, called DECADES (Database and Methodologies for Comparative Assessment of Different Energy Sources for Electricity Generation) [44].

The overall objective of the DECADES project is to enhance capabilities for incorporating economic, social, health and environmental issues in the comparative assessment of electricity generation options and strategies in the process of decision making for the power sector. The project, established in 1992, was carried out jointly with 9 international organisations. The project based its assessment framework for comparing alternative electricity generation options on the full energy chain (FENCH) approach that incorporates the different steps and levels in energy chains contributing to electricity supply. The FENCH approach was adopted because, for the purpose of electricity system analysis, it provides a more than satisfactory estimation of the emissions and residuals while avoiding the additional complexity of LCA. Comparisons of results from FENCH and LCA studies have shown that there are no significant differences in the emissions and residuals from electricity generation chains as estimated by the two methods, with the possible exception of renewable energy chains, for which “secondary” emissions and impacts picked up by LCA may be significant. It should be noted, however, that issues of non-electrical energy uses and demand side management (DSM) often are treated more fully through LCA than by the FENCH approach.

#### *3.5.4.5 Environmental activities in uranium mining and milling*

An increased awareness has been observed in all countries of the need for environmental protection of all steps of the nuclear fuel cycle, but in particular for uranium mining and milling. A study [61] has been conducted by NEA and IAEA on this matter and has described the different activities and developments in order to restore these sites. No specific life cycle analysis or detailed assessment was undertaken within this study.

For several years, large programmes have been underway in several countries to clean up wastes from closed mines and mills. For new projects, managers must prepare and submit an environmental impact assessment to regulatory authorities.

The report concludes that proper management and guidelines have been put in place in the different countries with mining and milling activities. In addition, the environmental protection strategies for non-radiological substances released to the environment by nuclear fuel cycle facilities are generally the same as for non-nuclear industries. Radiation protection measures are taken and, in case of high-grade ore mines, new mining techniques (e.g. non-entry mining or where appropriate ISL) are applied. Engineered retention barriers have been put in place to limit the long-term environmental impact of the tailings as waste from mining and milling. Today, the preferred option is to return the tailings to the mine from which the ore was excavated, where appropriate.

#### *3.5.4.6 Radiological impacts of spent fuel management options – the PARCOM study*

The NEA launched a comparative study on radiological effects of spent fuel management options in 1996 [62]. The main objectives of the study were to compile data and information on radioactive releases from various steps of nuclear fuel cycles, to analyse radiological effects in a systematic manner, to present results and scientific and technological interpretations, and to contribute to informed discussion in its Member countries. The study has been carried out by an ad hoc Expert Group on Spent Fuel Management Options under the Committee on Radiation Protection and Public Health.

Two reference PWR-based fuel cycles (i.e. open and closed) were selected to facilitate comparison of radiological impacts. Actual data on radioactive releases from reference facilities are used. Reference facilities are selected taking account of their scale, duration of operation, technological characteristics, and characteristics of the installed waste management process. Radiological effects were evaluated with a generic model.

One of the main findings of the study is that radiological impacts brought about by the two fuel cycle options considered are of about the same order of magnitude for both the general public and the fuel cycle facility workers (see Table 3.6). It should be remarked that electricity production in the nuclear power plant is the main source of the collective dose to workers and accounts for about one third of the collective dose to the public. The exposures in the mining and milling stage are caused by the daughter nuclides of the naturally occurring uranium decay chains. Collective doses are composed of very small doses to a large number of people over a long period of time. Each contribution is negligible compared with the level of natural background radiation. Much of the collective dose is produced by  $^{14}\text{C}$ . It should be remarked that only the collective dose to workers is considered appropriate in this context, as the concept of “collective dose to the public” may lead to wrong conclusions if not applied appropriately. Only the collective dose to workers is therefore considered in the list of indicators.

### 3.5.5 National studies

A study on trends in the nuclear fuel cycle, seen from a sustainable development perspective, could also be of value in defining the longer term R&D strategy by ensuring that there are a sufficient number of credible options available at the time when decisions about energy sources have to be made [63]. It is essential when making these decisions that the widest possible selection of the best technologies is available for consideration. This should be done in an environmental impact assessment (EIA) at the level of plans, policies and programmes. In such an EIA process all relevant technical, economic, environmental and social aspects should be identified and examined to create the best possible decision basis. The input to this selection process would demand information on relevant developments and trends, and on indicators which are considered to be relevant for use in the EIA.

Table 3.6 Summary of estimated doses for major fuel cycle stages of each option

| Fuel cycle stage                                      | Generic analysis,<br>Collective dose to population of<br>Europe, truncated at 500y<br>(manSv/Gwe-year) |             | Collective dose to workers<br>(manSv/Gwe-year) |             |
|---|--|-------------|--|-------------|
|   | Once-through   | Recycle     | Once-through                                   | Recycle     |
| Mining and milling                                    | 1  | 0.79 (1)    | 0.7  | 0.55 (1)    |
| Conversion, enrichment                                | 0 (2)  | 0 (2)       | 0.02   | 0.016       |
| Fuel fabrication                                      | 0.0009 (4)   | 0.0007 (3)  | 0.00657 (5)                                    | 0.0941 (3)  |
| Power generation                                      | 0.65 (6)   | 0.65 (6)    | 2.7 (7)  | 2.7 (7)     |
| Reprocessing,<br>vitrification and interim<br>storage | 0  | 1.534 (8)   | 0  | 0.012 (9)   |
| <b>TOTAL</b>  | <b>1.65</b>  | <b>2.97</b> | <b>3.43</b>                                    | <b>3.37</b> |

Remarks:

- (1) Scaled for recycle option based on the need for  $U_{\text{nat}}$ ; dose to workers taken from UNSCEAR88.<sup>38</sup>
- (2) Public doses included in contribution by fuel fabrication.
- (3) For recycle option weighted by  $\text{UO}_2$  and MOX fuel amounts (21.1 t & 5.5 t).
- (4) Public: specific analysis for Romans  $3.21\text{e-}4$ , Melox  $2.51\text{e-}3$ .
- (5) Workers: Romans  $6.57\text{e-}3$ , Melox  $4.3\text{e-}1$ .
- (6) Public: coastal 0.54, inland 0.65.
- (7) Workers: average for French 900 MW(e) units.
- (8) Public: generic analysis.
- (9) Workers: La Hague data.

38. New data are available in the UNSCEAR 2000 report.

### 3.5.5.1 France

A working group comprising experts from COGEMA, CEA, EdF and Framatome, performed an assessment of the different back-end fuel cycle options in France [64]. This study covered a rather detailed overview of the past decisions and possible options in the fuel cycle and assessed those according to a set of five criteria:

- Technological maturity.
- R&D needs.
- Investments needed for industrial, commercial realisation.
- Quantities of spent fuel, stocks, ...
- Flexibility of a certain option (scenario) to be changed towards another option according to the market conditions.

The Charpin-Dessus-Pellat report in 2000 [9] gave an overview of the main issues for the French nuclear power sector and emphasised especially the economic dimension. Several prospective scenarios for energy production in France were analysed for the time-period 2000-2050. The analysis indicated that nuclear energy is a competitive option and that the internalisation of externalities favours those scenarios having a slow increase of electricity demand and including a significant nuclear energy production. The analysis also showed that, due to the competitive generation cost for nuclear, any decision to phase-out nuclear would induce increased electricity costs. The decision whether or not to continue with reprocessing has only a minor impact.

### 3.5.5.2 GaBE project (Switzerland)

The Project GaBE is an ongoing Swiss research project of the Paul Scherrer Institute (PSI) in collaboration with the Swiss Federal Institute of Technology Zurich (ETHZ) [65]. The objective of GaBE is to develop, implement and use a comprehensive methodology for the consistent and detailed assessment of energy sources of interest under Swiss conditions and, thereby, to provide scientific support to decision making concerning the future configuration of the Swiss energy sectors. The GaBE interdisciplinary approach addresses the following aspects associated with different energy systems:

- Environmental impacts of normal operation.
- Health impacts of normal operation.
- Risk of severe accidents.
- Economics.

This approach allows the different aspects to be treated in an integrated manner, thereby making it possible to carry out comprehensive comparative studies of energy options, covering electricity, heating and transport systems. Figure 3.2 illustrates the assessment methods used and the basic interactions between the various modules. The overall approach is process-oriented, i.e. the technologies of interest and their features are explicitly represented, thus enabling a straight-forward accounting for technological improvements. Continuous lines reflect hard links, i.e. primarily data flows; dotted lines imply that some specific principles used in one approach are employed also in other module(s).



## **4. RESEARCH AND DEVELOPMENTS IN THE FUEL CYCLE**

### **4.1 Introduction**

Even though nuclear energy already is a proven reliable, economic and safe energy source, a continuing effort is made by governments and industry for its further improvement as a secure energy generation option. In general, while the balance in R&D funding between government and industry has changed over the past years, the goals of government and industry have remained fairly stable. The short to medium term economic aspects remain the primary interest of industry whereas governments focus primarily on the longer term aspects, e.g. sustainability. Chapter 2 introduced the fuel cycles operated today and gave an insight on the current status and constraints to the industry. This chapter will focus on those developments in the nuclear fuel cycle which contribute to improve the potential of the nuclear option as part of future energy generation portfolios. By showing the impact of these schemes (validated or estimated) on the indicators, introduced in Chapter 3, this and the following chapter aim at illustrating the added-value these developments could provide as well as the relative effort they would require.

While this chapter will describe the main technology developments within the separate steps of the nuclear fuel cycle, the reader is referred to Chapter 5 for a more comprehensive overview of the whole fuel cycle.

### **4.2 Developments in today's fuel cycle**

#### ***4.2.1 Developments in mining and milling***

Main developments:

- Operational improvements.
- Change in mining methods.
- Environmental restoration.
- New resources.

In a market characterised by a single-use product and, for the foreseeable future, by over-capacity, the mining and milling companies have embarked on cost-reduction and consolidation programmes in order to remain in the market. In addition to these competitive pressures towards cost-reduction, the operational improvements in mining and milling have been focused also on mitigating the environmental impacts. Increasingly, many countries require that environmental impact assessment (EIA) be conducted prior to development or extension of proposed uranium mining and milling operations. Research has been carried out by the uranium industry to develop operating strategies that meet these regulatory requirements, limit long-term impacts on the environment, demonstrate that facilities can be successfully decommissioned and returned to green-field status, and show that waste sites can be closed safely and the original environment can be restored [61].

The main long-term environmental issue associated with uranium mining and milling is the effective isolation from the accessible environment of the daughter-products of natural uranium (mainly radon decay products) that could migrate from uranium mill tailings into life-support systems and food chains. Research is being conducted on the option of isolating the remaining long-lived radio-nuclides ( $^{226}\text{Ra}$  and  $^{230}\text{Th}$ ) from the environment by disposing the tailings in the exploited mine pits and/or covering the tailings, or by separately disposing the radio-nuclides in a repository.

In order to prevent undesirable constituents in the tailings from leaching into the environment, engineered containment facilities are constructed for the long-term disposal of the material. While the 1993 UNSCEAR study [17], based on calculations for a theoretical site, indicated an annual collective dose from radon in uranium mill tailing areas to be 150 manSv per GWe, a more recent study [18] with real data on radon release rates<sup>39</sup> indicated a much lower annual collective dose of 1 manSv per Gwe.<sup>40</sup> This shows, next to the difference in assessment between both studies, that developments by industry were very effective in reducing every potential burden of this fuel cycle step. In addition, in mining operations where the uranium content of ore is high, the mining techniques and operating procedures are selected to reduce and control the radiation exposure of workers. This is true in particular for underground mines. In some cases, non-entry mining methods may be required. In this type of operation, remotely controlled equipment operated by personnel located outside of the ore zone is used to extract the ore.

As no ores are being brought to the surface during mining by *in-situ* leaching (ISL), concerns relating to radiation protection measures and waste management are greatly facilitated. Surface disturbances are minimised (there are no tailings) and therefore environmental impacts are much less than for conventional mining. The most important operational considerations are for the limiting of routine in-plant radiation health effects, as well as monitoring and controlling ground water conditions in and around the mined out ore body.

In addition to these operational developments in mining and milling, developments in search of new uranium resources for the very long term are ongoing. While the R&D budgets for uranium exploration have been reduced in the past decade [15], some efforts are continuing in the field of uranium extraction from seawater. If such extraction would become industrially feasible, the uranium contained in the world's oceans would make nuclear power an almost inexhaustible source of energy. The uranium concentration of seawater is low (approximately 3 ppb) but the quantity of contained uranium is vast – some 4 billion tonnes (about 700 times more than known terrestrial resources recoverable at a price of up to USD 130/kgU). Even if half of this resource could ultimately be recovered, it could support 3 000 GWe of nuclear capacity<sup>41</sup> for 6 000 years.<sup>42</sup> Research in Japan has shown that U-extraction from seawater would be technically feasible at a cost between about 300 and 700 USD per kg of  $\text{U}_3\text{O}_8$  [4]. Although this is 10 or 20 times the current uranium price, the overall impact on energy costs would be no more than about a 50 to 100% increase in the cost of electricity generated by nuclear power, owing to the fact that the cost of uranium at present contributes only about 5% to the cost of nuclear electricity generation.

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39. For real mining sites which comprise 80% of the present world uranium production.

40. This study is not using the concept of collective dose to the public, as the interpretation of this concept may be misleading. Only the collective dose to workers may be applicable.

41. Today's nuclear capacity is 373.8 GWe [Nucleonics Week, January 2001].

42. Assumes 75% capacity factor and next-generation reactors (e.g. high-temperature gas-cooled reactors) operated on once-through fuel cycles.

#### 4.2.2 *Developments in conversion and enrichment technology*

Main developments:

- Impact of market factors.
- Ultra-centrifuge and laser enrichment developments.
- Impact of new fuels on enrichment demand and characteristics.

Due to the increasing impact of several market mechanisms (i.e. evolution from an inventory driven market to a production driven market, overfeeding mechanisms by enrichers and the introduction of a tails re-enrichment market), developments in conversion and enrichment are oriented mainly towards cost reduction efforts and it is hoped that in the long term new facilities, based on advanced processes, could achieve costs of about 50 USD/SWU, if successfully developed. Most of the existing technology has been introduced in the 1950s, or even before, and continuous progress has been made in improving the conversion and enrichment processes. Enrichers using centrifuge technology have been increasing steadily their market share and show a good capability for adapting to market conditions.

Developments in the enrichment process have the potential to significantly change this fuel cycle step. Depending on the technology used, these developments could impact the conversion process and could significantly lower the cost of enriched uranium production. Advanced enrichment processes either imply improvements in the centrifuge process or involve use of lasers to separate the desired isotopes. Table 4.1 shows a summary of the present status of R&D in uranium isotope enrichment around the world [67]. Still in the research stages, the laser processes have been investigated for considerable periods but have not yet been deployed for production. Being very selective, laser processes for the production of enriched uranium for reactor fuel takes only a few steps<sup>43</sup> and avoids the multiple cascade stages required in centrifuge or diffusion plants. These processes also have relatively high throughput, avoiding the large number of parallel trains required in centrifuge plants. This would make the enrichment plants far smaller as well as reducing electricity consumption (to levels comparable to gas centrifuge). Two candidate laser technologies have been in recent development, the AVLIS/SILVA process that uses metallic uranium feed and the SILEX process, which uses uranium hexafluoride as the feed.

##### *AVLIS/SILVA*

- Atomic Vapour Laser Isotope Separation (AVLIS) is a high temperature atomic photo-ionisation approach to enrichment. In AVLIS, precisely tuned laser light and uranium vapour are brought together in a vacuum separator assembly. In the separator, atoms of the <sup>235</sup>U minor isotope in an atomic vapour stream of natural isotopic composition are selectively excited and photo-ionised by laser light. The selectively ionised atoms are then collected to generate a product enriched in <sup>235</sup>U. Once enriched, the metal product is purified and processed into an oxide suitable for fabrication into nuclear fuel. The use of metallic uranium in the enrichment process would impact also on the previous conversion process. The commercial benefits of this laser enrichment technology

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43. The enrichment factor of an isotopic separation process is defined as the relation between the isotopic ratios in the product and in the feed materials ( $\alpha = R_p/R_f$ ) in one stage. In the AVLIS process, this ratio lies between 7 and 200. These values are much higher than those reached by each stage of gaseous diffusion ( $\alpha = 1.002$ ) and ultracentrifugation ( $\alpha < 1.3$ ) methods. This fact allows the necessary enrichment of uranium for nuclear reactors in only one stage. Moreover, it allows the production of enriched uranium through “tails stripping”, that is, by re-enriching the depleted uranium (tails) remaining after uranium enrichment by diffusion or centrifuge processes.

therefore cannot be seen as simply applying to SWU, but rather must be viewed from the broader perspective of the total fuel cycle.

- Laser enrichment advocates were disappointed when the U.S. Enrichment Corporation (USEC), after its privatisation, terminated AVLIS deployment, despite the construction of a large pilot plant and sunk research budgets of about USD 1.3 billion by the U.S. government, and about USD 500 million by USEC, which spent annually about USD 100 million in the AVLIS research programme. The reason given for the termination was that the projected returns were not sufficient to outweigh the financial risks and ongoing capital expenditures imposed upon USEC to construct and operate an AVLIS plant. The estimated construction costs for a new AVLIS plant were beyond USD 2.5 billion for an annual capacity of 9 million SWU. Additionally, the deterioration of market conditions (lowering of SWU and uranium selling prices) significantly contributed to the decline of projected returns on investment from AVLIS deployment.
- In France, the SILVA programme progressed, while CEA and Cogéma continued to evaluate the role of SILVA in light of the AVLIS programme termination. The decision to proceed with SILVA demonstration and deployment depended heavily on the French assessment of the availability of low cost power for its gas diffusion plant and on the prospects for continued technical improvement in SILVA. Estimates by Urenco showed that further financial commitments would be in excess of USD 1 billion in order to deploy AVLIS/SILVA technology [69]. In January 2001, CEA announced that they were terminating the development of this technology.

#### *SILEX*

- USEC continues to support financially the laboratory scale development of the Australian molecular laser enrichment SILEX<sup>44</sup> (Separation of Isotopes by Laser Excitation) concept. This process claims the theoretical advantage of being simpler and less expensive than AVLIS, since SILEX is a UF<sub>6</sub>-based process that would not involve changes to conversion processes upstream and downstream of the enrichment step. SILEX, using UF<sub>6</sub>, avoids the corrosive AVLIS environment with high temperature vapour and liquid uranium metal. In SILEX, lasers selectively excite <sup>235</sup>U-containing UF<sub>6</sub> molecules, which are collected to yield a product enriched in <sup>235</sup>U.
- The SILEX development programme includes characterisation, direct measurement and pilot plant phases. The characterisation phase, which measured the key physics parameters that characterise the enrichment was completed successfully in January 2000. Confirmation of these parameters is one of the steps in the direct measurement phase that will demonstrate whether there is a reasonable potential for the separation of commercial quantities of enriched uranium. The next phase will lead to the design of a pilot plant and allow more detailed calculations of the economics of the SILEX process.

#### *Gas Centrifuge*

- The United States is evaluating whether to initiate a new programme to develop an advanced gas centrifuge as a means to assure the future domestic supply of enriched uranium. Building on earlier development programmes, the United States plans would develop and deploy a pilot-scale gas centrifuge demonstration plant on the site of the Portsmouth Gaseous Diffusion Plant. Programme goals are to design a 325 SWU per year machine and operate a 240-machine pilot plant for a duration sufficient to verify the reliability and commercial potential of an advanced gas centrifuge design.
- In Japan, advanced centrifuge development continues at the Japan Nuclear Fuel Development Institute (JNC). Advanced centrifuge development is carried out by JNC and was adopted in the

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44. Also abbreviated as MLIS, **M**olecular **L**aser **I**sotope **S**eparation.

Rokkasho plant. In order to improve the economy of the plant, JNC developed a high-performance centrifuge with new composite materials. Throughputs of this technology were claimed to achieve 50 SWU/year per machine, as compared to early centrifuge models which typically produced 2-5 SWU/year per machine.

- In general, it should be noted that technical developments in other fields (lower-cost high-strength fibres, solid-state motor controls, computational fluid dynamics, ...) have also continued to improve the economics of centrifuge enrichment and makes it the preferred state-of-the-art enrichment technology. Except for the specific case of re-enriching reprocessed uranium, for which laser enrichment processes would be more suitable, the future enrichment market will likely be increasingly dominated by centrifuge enrichment plants.

Table 4.1 **Technology status<sup>45</sup> in uranium enrichment** [excerpt from Ref. 68]

| Country                   | Diffusion | Centrifuge | Chemical | Jet Nozzle | Laser | Plasma |
|---------------------------|-----------|------------|----------|------------|-------|--------|
| Argentina                 | II        |            |          |            |       |        |
| Australia                 |           | I          |          |            | I     |        |
| Brazil                    |           | II         |          |            | I     | I      |
| USA                       | IV        | II         |          |            | II    |        |
| France                    | IV        | I          | II       |            | II    | I      |
| Netherlands               |           | IV         |          |            | I     |        |
| Italy                     | I         | I          |          |            | I     |        |
| Japan                     |           | IV         | II       |            | I     |        |
| Pakistan                  | III       | IV         |          |            |       |        |
| Germany                   |           | IV         |          | II         | I     |        |
| China                     | IV        | IV         |          |            | I     |        |
| Russia                    | IV        | IV         |          |            |       |        |
| Sweden                    |           |            |          |            | I     |        |
| UK                        |           | IV         |          |            | II    |        |
| General Development Stage | IV        | IV         | II       | II         | II    | I      |

#### 4.2.3 *Developments in fuel design and fabrication*

Main developments:

- Standardisation in fuel assembly components.
- High burn-up fuels.
- Inert matrix fuels.
- HTGR fuels.

While the front-end fuel cycle steps are technologically rather independent from the choices made in the back-end of the fuel cycle, the same does not apply for the fuel design and fabrication step. Fuel design and fabrication of fuel are very important elements of any fuel cycle as the fuel assembly not

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45. Throughout this chapter, the following keys are used to indicate technology status:  
I = Research Stage; II = Laboratory/Pilot Plant Stage; III = Pre-industrial Stage; IV = Industrial Stage.

only fulfils the role of interface between the fuel cycle and the nuclear power plant, but also impacts the options in the back-end of the fuel cycle.

Several developments are ongoing to improve the interface between the fuel cycle and the power plant as well as to develop additional options for the back-end of the fuel cycle. Most of today's development activities on fuel are conducted by the nuclear industry as a continuous improvement process aimed towards higher performance (burn-up, fuel utilisation), higher reliability ("zero-defect fuel"), and higher cost-effectiveness for the fabricator as well as for the users. Development activities related to the safety of the fuel and the power plant are undertaken by governmental institutes and through international co-operation, e.g. OECD/NEA Halden reactor project. New fuel types, such as nitride fuel, metal fuel and inert matrix fuel, are being studied by research organisations in the framework of long-term options for the nuclear fuel cycle.

To meet society's request for protection of the environment, environmental protection measures are introduced in the fuel assembly fabrication line (or process), e.g. by introduction of a dry conversion process, completely avoiding production of any wastes and allowing for sale of by-products (HF, CaF<sub>2</sub>).

#### *4.2.3.1 Fuel developments in the short term*

Over the next several decades, it is envisioned that oxide fuel will remain the standard, and will constitute the bulk of irradiated fuel arising from LWRs. Due to the conservative nature of utilities in the short-term in the light of the high costs and risks faced in attempting any changes to the operating licences of their capital-intensive nuclear facilities, there is considerable institutional pressure to retain this fuel type. Developments are therefore mainly incremental, aiming at improving the performance and reliability of the fuel and its cost-effectiveness.

Improvements in the performance, reliability and cost-effectiveness of nuclear fuel for the present reactors are a continuous process, which is undertaken mainly by the fuel fabricators. The deregulated and increasingly competitive market for the electricity utilities has put more emphasis on increased burn-up of fuel, flexibility in cycle length, up-rating the output of power plants, and in general a higher reliability of fuel. In France, for example, EdF has shown a fuel cycle cost reduction of 9% by changing from a 1/2-core management to a 1/3-core management. In addition, changing from 1/3-core to 1/4-core and from 1/4-core to 1/5-core management reduces the fuel cycle costs by 6% in both cases. This change of core management is associated with an increase of fuel burn-up [69]. Standardisation of fuel assembly components has been introduced gradually in order to enable flexibility in fuel assembly composition and shorter delivery times while securing cost-reductions for the fuel fabricator as well as for the end-user. It should be remarked, in this context, that most utilities have changed their purchasing policy by reducing their stocks of high-value commodities (e.g. fabricated fuel assemblies) in favour of low-value commodities (e.g. natural uranium).

In the future, the choices made in the early years of nuclear deployment will gradually change to integrate more a life-cycle design approach in the fuel specification and fabrication. This would, however, demand more than just incremental developments. Over recent years, a number of plants have suffered de-ratings, or even premature termination of an operating cycle, due to unanticipated problems associated with the fuel. Many of these problems can be associated, at least in part, to increased duty on the fuel, i.e. excessive corrosion, significant secondary degradation of the cladding following a primary failure, incomplete rod insertion, fuel failure due to grid-rod fretting, axial offset anomalies, and others. In these cases, nuclear plant operators are reviewing their specific situation and balancing the benefits of lower fuel costs resulting from increased fuel duty against the risks of increased fuel failure. This, combined with reduced R&D expenditure by the fuel vendors, indicate that in the longer term, a number of co-operative industry initiatives will be needed to focus on

answering the open questions on fuel performance. The extent to which these partnerships are formed and the success they achieve likely will determine the scope and pace of continued advances in oxide fuels.

Within the scope of the research on safety, the high temperature behaviour of ceramic (SiC) encapsulated UO<sub>2</sub>-pellets and absorber elements are presently investigated by German research centres. The topic fits into the research investigations on the use of ceramic materials in innovative light water reactors and contributes to the development of both a new fuel element and new absorber element with fundamentally higher lifetimes in the high temperature range. Fundamental questions on the suitability and the material behaviour of SiC have been investigated already in research projects. Based on the results, the current work deals with the detailed analysis of the interactions between the SiC and UO<sub>2</sub>, Zircaloy steel, absorber material, oxidising steam, non-oxidising or inert atmosphere. The high temperature ranges, even above 2 000°C, will be investigated using the test facilities OSIRIS and DRESSMANN.

#### 4.2.3.2 *Advanced fuel concepts in the longer term*

Innovative alternatives being investigated for the future development of nuclear power, and especially the fuel in such schemes, aim toward addressing some of the major issues raised above. Future directions in optimisation of the fuel cycle will have to address the integration of front-end and back-end issues together with the operational performance of nuclear power plants. In that respect, innovative fuel forms have been proposed and studied in view of improving resource conservation and waste management. These fuel forms, which satisfy the newer requirements and improve or at least do not penalise the economy of the fuel cycle, have yet to be deployed but current investigations and experiments have led to some promising developments. The time required to effectively perfect a new fuel, which entails experimental irradiation tests,<sup>46</sup> is about 20 years, putting the maturation of these developments around 2020.

The increase of fuel burn-up and the consequent increased accumulation of minor actinides and fission products in the fuel result in more severe constraints on the fuel itself. Additional constraints on fuel behaviour are introduced by options in the back-end of the fuel cycle depending on each country's nuclear policy, e.g. long interim storage, reprocessing, recycling. In general, two main options are envisioned to categorise the innovations in fuel technology:

- In once-through scenarios, higher burn-ups are achieved in combination with defect-free long-term immobilisation of the radioactivity in interim storage or in geological disposal. In these options, oxide fuels are further developed to achieve higher burn-ups and inert, rock-like fuels are considered to allow incorporation of excess plutonium coming from civil or military stocks for once-through irradiation.
- In recycling scenarios, actinide recycling requires high burn-up fuels (especially in fast reactor conditions) containing high loading of plutonium and minor actinides. Multiple recycling adds further constraints on the potential to reprocess these highly active fuels. In a homogeneous recycling scheme,<sup>47</sup> MA-containing oxide fuels are researched where, for Pu contents above about 40%, nitride fuels with enriched <sup>15</sup>N support are considered. Metal fuels

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46. It should be remarked that the number of available experimental irradiation facilities is decreasing and even improved simulation codes only partially compensate for this loss in experimental verification capability.

47. In a homogeneous recycling scheme, the transuranic nuclides are not separated from each other for recycle in the reactor. A heterogeneous recycling scheme demands the separation of Pu, Np, Am and Cm in order to recycle those as dedicated targets in the reactor.

are considered in more “integrated” types of fuel systems. In case of heterogeneous recycling of Pu and MAs, inert matrix fuels, e.g. CERCER/CERMET,<sup>48</sup> are considered.

The difference in objectives for each of the above options indicates an interest to engage in a life-cycle design philosophy for new fuels. Although the intent and the technological targets of the two fuel options appear to be diverse, the technical issues are, in fact, strongly connected to each other. High burn-up, i.e. above 70 GWd/t, and actinide burning both demand particular attention to fission gas release, cladding corrosion, cladding ductility, structural strength and radiation growth (swelling). New fuel materials are being reviewed which could increase the fuel loading while also increasing the fuel’s thermal conductivity and allowing improved fission gas retention. Use of inert matrices for fuels to increase the burning rate of actinides, “integrated” fuel types such as metallic fuel for on-site reprocessing (IFR-type) or fuels mimicking natural minerals are developments useful in both fuel options. Inert matrix fuels (IMF) are being developed for use in both LWRs or FRs. More advanced fuels, e.g. to incorporate higher Pu contents, are of nitride rather than oxide or even metallic form for more “integrated” type of fuel scenarios. One specific variant (see section 5.3.2), referred to as the DUPIC fuel cycle, relates to a mixed use of LWR and CANDU reactors, where LWR spent fuel is reused in CANDU reactors, demand reprocessing, without Pu and U separation, followed by refabrication of fuel.

Due to the delay in FR programmes, Pu stocks have increased and those countries that embarked on a recycling strategy have begun to focus on consuming this Pu in LWRs. While MOX technology has matured, present licensing constraints limit the ability of MOX loaded LWRs to decrease the Pu inventory.<sup>49</sup> Therefore, increasing the consumption rate of Pu in LWRs by an increase of the total amount of loaded Pu is mandatory if countries desire to decrease their plutonium stocks. The main limitations towards 100% MOX in standard LWRs are related to the coolant void effect and to core control in normal or transient conditions.<sup>50</sup> One solution would be to use enriched boron in soluble boron and shutdown rods. Calculations performed with a 100% MOX fuelled core indeed show acceptable characteristics and cooling transient behaviour, but the reactivity margins are not large enough to allow multiple recycling and large burn-up. It is clear that to go further in terms of plutonium consumption without compromising on safety guarantees, more advanced fuel and reactor concepts are needed:

- High Moderation Reactors (HMRs) that correct the negative effects of higher Pu contents on the safety characteristics of standard LWRs.
- Diluting the plutonium to low concentrations in UO<sub>2</sub>-fuel, i.e. MIX fuel cycle concept.

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48. Developments being pursued related to a recycling scenario, e.g. CERCER and CERMET, have comparable analogues applicable to once-through scenarios, e.g. ROX-type fuels. Most of the advanced fuel concepts still rely on oxide fuel form.

49. LWRs are limited to 30% of the core being MOX fuel because of the effect that the plutonium has on reactor physics and safety limits in existing standard LWRs, e.g. time-dependent safety limits, Doppler broadening effects, reactor kinetics, and the delayed neutron fraction. In Japan, BWRs have been licensed for 100% MOX core.

50. In case of voiding, the neutron spectrum becomes faster, the neutron flux in the thermal region tends towards zero and is concentrated in the region from 10 keV to 1 MeV. Thus, all the captures by <sup>240</sup>Pu and <sup>242</sup>Pu in the thermal and epithermal resonance disappear and the <sup>240</sup>Pu and <sup>242</sup>Pu contributions to the void effect become positive. Finally, the higher the Pu content and the poorer the Pu quality the larger the void effect. Concerning core control, Pu enrichment leads to decrease in  $\beta_{\text{eff}}$ , the efficiency of soluble boron and control rods. In addition, the Doppler effect tends to decrease when Pu replaces U, so that in case of transients the core could diverge again if the control is not effective enough. As for the voiding effect, the plutonium degradation and the <sup>240</sup>Pu and <sup>242</sup>Pu accumulation after multiple recycling lead to spectrum hardening and to decrease in control.

- Use of Inert Matrix Fuel (IMF) assemblies in standard LWRs, e.g. French PLUTON concept where uranium-fuel is replaced by IMF (or by thorium).
- Some more specific fuel concepts such as the French CORAIL and APA<sup>51</sup> concepts where some standard UOX fuel pins are replaced by standard MOX fuel pins (a kind of heterogeneous MIX concept) or annular uranium-free fuel pins.
- Reduced-Moderation Water Reactors (RMWRs) have been studied recently in Japan as one of the advanced water reactors. The reactors are based on LWR technologies and are expected to have high conversion ratios around 1.0 using MOX fuel. The concept of the reactor has been developed at JAERI [70,71] to realise favourable features such as the effective utilisation of uranium resources, high burn-up and long operation cycle, and the multiple recycling of Pu.

Analyses of these concepts indicate that the HMR, RMWR and MIX options present effective opportunities to increase Pu consumption in LWRs. The safety coefficients are globally acceptable and very close to that of a standard LWR fuelled with UOX. On the contrary, the PLUTON concept does not present sufficient safety margins as regards the coolant void effect and is not really feasible in a strategy of multiple recycling. However, it retains potential in the case of a once-through strategy. The APA concept represents acceptable features. Some improvements should be obtained through optimisation of the design, but both these concepts are unsuitable in a scenario that recycles plutonium more than once or twice [72], as degradation of the plutonium quality by multiple recycling greatly degrades core performance.

The feasibility of indefinite plutonium recycling using a few LWRs (about 20% of a nuclear park) has not been demonstrated yet. Even the best concept, HMR, designed to enhance the moderation ratio by a factor of 4, does not reach suitable coolant void parameters<sup>52</sup> in a multiple recycling strategy.<sup>53,54,55</sup> However, this should be viewed also from the perspective that such Pu burning scenarios

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51. Advanced Plutonium fuel Assembly.

52. A parametric study carried out to assess the effect of the moderation ratio demonstrates the strong impact of the reactivity coefficients. For example, in a 100% MOX core, the increase of the moderation ratio from 2 to 4 brings about an increase of the boron efficiency by a factor 2.5, and a 20% decrease of the Doppler coefficient. The moderator temperature coefficient becomes less negative as well. The main cause is the difference in the Pu inventories: increasing the moderation ratio provokes a 60% decrease of the Pu mass in the assembly (for a similar fuel management). Furthermore, the moderation ratio has to be as high as possible to improve plutonium consumption and reduce minor actinide production.

53. Compared to a standard UOX-LWR case, the higher plutonium content leads to an increase of minor actinide production by a factor of 7, while the radio-toxicity reduction factor due to the Pu-recycling only reaches about 3 to 5. In addition, standard 17x17 type assemblies cannot be used due to limitations on the surface heat flux of the fuel rods. Turning to the 19x19 type of assemblies, keeping the same external dimensions of the fuel assemblies, is therefore necessary and demands changes to the reactor core geometry. Some detailed studies [73] indicate that the plutonium consumption and MA formation for a fuel cycle using HMR would become  $\Delta\text{Pu} = -74.2 \text{ kg/TWhe}$  and  $\Delta\text{MA} = 9.5 \text{ kg/TWhe}$ , based on a 6.6% initial Pu content. Other fuel management approaches, with increasing Pu-content up to 14.4%, indicate a  $\Delta\text{Pu} = -134 \text{ kg/TWhe}$  but also a higher increase of MA production to 11 kg/TWhe.

54. From a fuel fabrication point of view, fabrication and handling of neptunium contained at 2 wt% level in typical LWR-MOX fuel would not pose specific problems. In a MOX fuel factory, with respect to the pure  $\text{PuO}_2$  reference case, the presence of  $\text{NpO}_2$  does not affect the alpha and neutron emission but increases the gamma source, due to  $^{235}\text{Pa}$  (daughter of  $^{237}\text{Np}$ ). The powder blending glove box should be protected by some 2 mm-thick Pb layers to keep external dose rates unchanged [75]. In case of  $^{237}\text{Np}$  multiple recycling, the content of  $^{238}\text{Pu}$  increases progressively, and this additional source of neutrons and heat affects further reprocessing and refabrication steps. Np should therefore be irradiated preferably in fast reactors with a

in LWRs would demand a significant increase in MOX fuel fabrication capacity (especially in the MIX concept). For example, a nuclear park of 60 GWe producing 400 TWhe/year would require fabrication of 200 tonnes of MOX fuel annually, which corresponds to about 25% of the fabrication plant capacity being dedicated to MOX technology. This indicates that fabrication and handling of such highly-active fuels, especially in the case of advanced partitioning and transmutation cycles, should be in concentrated, lower throughput facilities in order to keep safety compliance and costs under stringent control. Thus, despite the excellent performance of MOX fuel use in today's LWRs, this fuel cycle option should be considered only as an intermediate solution before Pu would be used in fast reactors.

A fast reactor (FR) has advantages compared to LWR in the utilisation of MOX fuel due to the FR core physics characteristics: it is able to reach higher burn-up; can use fuel with higher Pu content; and can accept higher contents of minor actinides and FPs. This means that a FR provides an effective resource utilisation and transmutation of radio-toxic nuclides. MOX is a well-developed fuel form for FRs, but the fuel economy needs to be improved for their commercialisation. Developments aim primarily at the realisation of high burn-up fuel for cost reduction. This requires the development of advanced cladding to resist fuel swelling and mechanical load at high MOX burn-up (e.g. 150 GWd/t) or high neutron fluence (e.g. 250 dpa<sup>56</sup>). JNC, for example, has developed oxide dispersion strengthened ferritic/martensitic steel as the most promising cladding material. JNC also is conducting development of simplified pelletising methods, which eliminate complicated processes such as uranium/plutonium content adjusting, sintering, etc., and which leads to a more compact fabrication facility. Vibro-packing, which Russia, Switzerland and other countries have developed, is another attractive option because of its very simple process. Nitride fuel and metal fuel have been studied in many countries, as they have excellent neutron economy in the core, thereby offering the possibility not only for economic improvement but also for transmutation of long-lived nuclides.

The burning of Pu in FRs has been extensively studied in France and Japan. More work is ongoing in relation to more advanced fuel cycles, i.e. partitioning and transmutation (P&T) fuel cycles in which the minor actinides are recycled also (in either FRs or ADS systems). Recycling of the minor actinides (americium and neptunium) is possible in fast neutron reactors, in both the homogeneous and heterogeneous modes. The mass balance shows the advantage of a fast neutron spectrum over a thermal spectrum in allowing a higher burn-up to be reached. However, there remain some issues that require resolution:

- Incineration of americium generates a large amount of curium that must be processed to diminish the overall radio-toxicity of the waste.
- Experiments performed in Superphénix (SUPERFACT) revealed considerable helium production in targets containing americium, which could limit the permissible concentration of americium in fuels. The incineration of curium should be considered as highly problematic due to its high neutron and alpha activity.

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lower capture to fission ratio and consequently result in a reduction <sup>238</sup>Pu yield from irradiation. In general, Np recycling in LWRs and FRs would not pose specific problems on the fabrication level.

55. The addition of americium to plutonium powder in a MOX fuel fabrication plant would induce a strong increase of the gamma-dose (by a factor of 4.5). This increase could be mitigated by the addition of shielding with a layer of 25 mm of steel. It thus appears that dose rate could be controlled at the expense of extra shielding, but this would hinder the fabrication and increase the cost of operations. Still, the operations seem to be a feasible extension of the standard MOX fabrication conditions, especially if one considers remote fabrication with automated processes.

56. dpa = displacements per atom.

- Separating the curium after removal from the reactor and placing it in interim storage for a sufficiently long period (about 100 years) should be considered as among the several options for handling curium. After this storage, the daughter product ( $^{240}\text{Pu}$ ) could be recycled in fast reactors. This solution, however, involves the interim storage of large quantities of a highly radioactive element, which will require further assessment from a safety standpoint in particular.

All these concepts need very specific fuels or targets, capable of attaining high burn-ups as well as being suitable for reprocessing in multiple recycling schemes. Most concepts for such fuels have been based on extension of today's standard oxide fuel experience and involve the use of inert matrix fuel concepts.

#### *4.2.3.2.1 Fertile fuels*

Fertile fuels, i.e. fuels including uranium or thorium as matrix, are developed for LWR and FR use. Depending on the fuel cycle scheme, different forms of fertile fuel, i.e. oxide, metal or nitride form, will need to be developed.

#### *Oxide fuels*

The first experience in fabricating oxide fuels with high contents of Am and Np (up to 20% Am) was achieved by the Institute for Transuranium Elements (ITU) in Karlsruhe (Germany) during the 1984-1986 period. The fabrication used a sol-gel process followed by pressing and sintering of the spherical particles. A systematic programme has been planned in JNC (Japan) for fabrication and investigation of irradiation behaviour of MOX containing MAs. Two fabrication methods, pellet-pressing and vibro-packing, have been studied for neptunium-based fuel pins. Irradiation of MOX fuel containing Np and Am is planned in JOYO, to be initiated around 2003.

#### *Metal fuels*

Since the 1960s, Argonne National Laboratory (USA) has been engaged in developing metal alloy fuels. For plutonium-containing fuels, U-Pu-10wt%Zr alloy was selected as having a high melting point and compatibility with stainless steel cladding. Since 1984, U-Pu-Zr alloy fuels have been further investigated as part of the IFR programme. More than a thousand fuel pins were fabricated by injection casting and irradiated in EBR-II, some of them to a maximum of 18 atomic % burn-up with cladding temperature  $<600^\circ\text{C}$  and linear heat rate  $<500\text{ W/s}$  without failure.

The multi-component alloys U-Pu-Zr-5MA-5RE and U-Pu-Zr-2MA-2RE were selected by CRIEPI (Japan) for irradiation study and metallographic characterisation. CRIEPI will plan irradiation tests of such fuels as soon as difficulties in restarting the Phénix reactor have been overcome.

#### *Nitride fuels*

Nitride has many advantageous properties as advanced fuel, such as high thermal conductivity, good FP retention, high heavy metal density and mutual solubility, and has the potential to be reprocessed by pyrochemical methods. In 1970s and 1980s, laboratory studies on a fabrication process for mixed nitride fuels were performed at CEA (France). The fuel was seen as the best advanced fuel for FR application, mainly due to its higher breeding gain and the absence of fuel/sodium reaction in the case of cladding failure. The process developed to produce the fuel was the carbothermic reduction of mixed oxide in an atmosphere of nitrogen, followed by decomposition of higher nitrides to mononitride. This dry preparation process was deemed the most suitable for industrial applications. Nitride pellet fuel pins have been fabricated for irradiation tests in the French Phénix reactor, and the ITU (Germany) fabricated nitrides by sol-gel process for irradiation in the HFR reactor. During the

past 10 years, research on fabrication technology for mixed uranium-plutonium nitride fuel has been performed by JAERI with a view to an advanced fuel cycle system. Nitride pellets have been fabricated and supplied for measurement of their characteristics and for irradiation tests. JNC has also fabricated some nitride fuel pellets and measured the fuel characteristics.

Today, pellet-type nitride fuel is fabricated by milling, compacting and sintering of particles obtained from the conversion of oxide to nitride by carbothermic reduction. No important problems appear to remain except in the enrichment and recycling of  $^{15}\text{N}$ . The use of  $^{15}\text{N}$  would be preferable because of the massive formation of  $^{14}\text{C}$  that results from the  $^{14}\text{N}(n,p)^{14}\text{C}$  reaction. Both  $^{15}\text{N}$  and  $^{14}\text{C}$  would be difficult to retain during the nitride dissolution in the PUREX reprocessing, while they could be recovered easily during pyrochemical reprocessing with fused salt electrorefining. In view of the cost of enrichment of  $^{15}\text{N}$ , which is only 0.365% in natural nitrogen, the recycling of  $^{15}\text{N}$  would be one of the major issues for the feasibility of nitride fuel. JAERI has recently shown the possibility of  $^{15}\text{N}$  recovery to a high yield by dissolving nitrides in the fused salts [75].

Nitride is suitable also as a target compound for burning americium by diluting it with inert materials such as ZrN and TiN. The fabrication of the solid solution of ZrN and UN (as a surrogate for AmN) by a sol-gel route has been studied. Such fuel is considered also to be a candidate for plutonium burning in the CAPRA reactor.

Fast reactors offer an effective solution to the issues involved with effective management of plutonium inventories. The superiority of the fast reactor, as Pu-burner or self-sufficient breeder, for managing plutonium of any isotopic grade is clearly demonstrated in several studies and full-scale experiments. Firstly, it is possible to increase the plutonium content of the fuel theoretically up to 100%, but more realistically up to 45%, without affecting unduly the safety parameters of the reactor. Secondly, the plutonium consumption, even if the Pu grade is poor, is accompanied by limited minor actinide production.

#### 4.2.3.2.2 *Inert matrix fuels*

Higher net burning rates for plutonium or minor actinides may be achieved by exclusion of fertile materials in the fuel, i.e. use of inert matrix fuels (IMF). These fuels are under development for use in LWRs as well as in FRs and for heterogeneous recycling schemes or for once-through irradiation schemes, e.g. if one wants to burn minor actinides.

Several groups are working on inert matrix fuel concepts. In France, the CEA is conducting research on ceramic-ceramic (CERCER) fuels, ceramic-metal (CERMET) fuels and solid solution pellet (SSP) fuels. Experiments have, however, indicated that CERMET fuel shows higher integrity than CERCER fuels, and also the release of  $^{85}\text{Kr}$  is three times lower in CERMET than in CERCER fuels. The advantage of CERMET fuel lies in the fact that the metallic matrix is a potential supplementary barrier to fission gas release and ensures excellent thermal conductivity, as well as forming a metal to metal contact for the cladding and matrix, which makes PCI highly improbable. With respect to CERMETS, the notion of such “cold fuel” (i.e. fuel with high thermal conductivity) offers enormous potential for investigation with a view to achieving very high burn-ups, as well as opening up new perspectives with regard to severe accidents. CERCERs need additional R&D in order to increase their resistance to irradiation and to increase their thermal conductivity and viscoplasticity [76].

Various lines of research (e.g. by JAERI, PSI, ENEA and CEA) are oriented towards a yttrium-aluminium-garnet (YAG) configuration with a view to obtain a “rock-like” matrix, which would be optimal for ultimate waste (spent fuel) disposal in deep geological formations. In order to be economically advantageous, this solution must enable high burn-up so that the final fissile nuclear inventory is as low as possible. However, the poor thermal conductivity of these materials is likely to create heat removal difficulties.

Another option for the disposition of excess plutonium, from both spent nuclear fuel and dismantled warheads, is a system of plutonium rock-like oxide (ROX) fuels and their once-through burning in LWR has been investigated, especially in Japan [78]. The ROX fuel consists of chemically stable fluorite phases (stabilised  $ZrO_2$  or  $ThO_2$ ) and spinel ( $MgAl_2O_4$ ) and the  $PuO_2$  is solidified in a fluorite phase. The zirconia will dissolve actinides and rare earths, while the spinel will fix fission products of alkali and alkaline-earth metals such as Cs and Sr. The spent ROX fuel also consists of natural analogous geologically stable phases, and would be disposed directly as high level wastes after about 50 years of cooling. The chemical stability of ROX in nitric acid would in addition provide a “chemical barrier” against reprocessing, thereby enhancing proliferation resistance. In all the stages of fuel fabrication, irradiation and disposal, the ROX-LWR system is built basically on conventional technologies and facilities. Committed individual dose equivalents from disposed spent ROX fuels are estimated to be lower than those from MOX and  $UO_2$  fuels by more than two orders of magnitude (cfr. SYNROC) [78]. However, the spent fuel disposal cost in the ROX-LWR systems is rather high compared to today’s estimates for LWR-UOX and MOX fuel cycles. Another obstacle relates to the very small Doppler coefficient, reactivity swing due to absence of any fertile nuclide and low melting temperature, thus making this fuel less favourable in LWRs and necessitating the addition of a fertile nuclide ( $^{232}Th$  or  $^{238}U$ ). Several improvements were studied to solve these difficulties of the ROX fuel. As a fuel material, spinel may have unfavourable irradiation effects such as swelling and, to avoid these irradiation effects, a fuel with fluorite particles dispersed in an inert matrix is being examined, together with a fuel having a two-phase homogeneous mixture of fluorite and spinel.

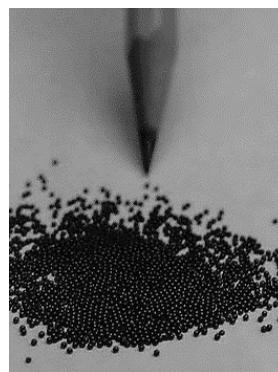
#### 4.2.3.3 High temperature gas reactor fuels

The basis for most of the HTGR-fuels is based on the TRISO (Tri-ISotropic) coated particle fuels containing a tiny spherical kernel of fissile or fertile material, as appropriate for the application, encapsulated in multiple coating layers (see Figure 4.1). A low-density carbon (buffer) layer surrounds the kernel to attenuate fission recoil atoms and provide void volume to accommodate fission gases. Changing the relative volumes of the kernel and buffer enables to accommodate different fuel compositions (e.g. UOX fuel, ex-weapons-Pu burning, ...). Surrounding the buffer in successive layers is an inner pyrocarbon (PyC) coating, a silicon carbide (SiC) layer, and an outer PyC coating. Taken together, these layers form a barrier that is highly corrosion resistant and essentially impermeable to the release of gaseous and metallic fission products. The overall diameter of standard TRISO coated particles varies from about 650 microns to about 850 microns. The fabrication of the spherical fuel particles (microspheres) for use in HTGRs has been accomplished in Russia, Japan, the Netherlands, Germany and the USA. Fuel microspheres are produced from uranium dioxide (with approximately 10%  $^{235}U$  enrichment) by sol-gel processes, slip casting methods and dry agglomeration. Comparison of the main geometrical and structural characteristics of fuel microspheres manufactured by these methods showed that sol-gel is the most acceptable method. After formation of a uranium nitrate broth, spherical droplets are produced by a vibrational dropping technique. The droplets are aged to improve the internal structure and then washed to remove the nitrate. After drying and calcining, the calcined kernels are reduced to  $UO_2$ , and kernel fabrication is completed by sintering to produce highly dense material. Coating of the kernels is performed in fluidised beds by pyrolysis of hydrocarbons for PyC and of silane derivatives for SiC. Because of criticality restrictions, the maximum batch size in the coating process is 1-10 kg of heavy metal oxide. Among the main specific requirements related to radiation performance of HTGR spherical fuel particles is that fission product release should be smaller than  $10^{-5}$ , which is equivalent to one damaged coated particle per 10 000 particles. Such stringent requirements can be met only when the starting product, i.e. the fuel microspheres, is highly homogeneous. Therefore, along with ensuring specified average values for the basic characteristics of the microspheres, a high degree of homogeneity is needed both within one batch and between batches of microspheres [79].

The TRISO coated fuel microspheres may then be included in, in principle, two types of fuel elements:

- As fuel pebbles (balls) for use in so-called pebble bed gas-cooled reactors. In this case, the microspheres are bonded into bigger fuel pebbles (“balls”, 6 cm in diameter). In the reactor, about 700 000 of these fuel pebbles make a loose bed in a cylindrical graphite structure measuring about 6m in diameter and in height (for instance the former 300 MWe THTR-300 plant). During operation, fuel pebbles are added continuously through openings in the reactor core top reflector and withdrawn through a central pipe in the bottom of the reactor core. The fuel pebbles pass repeatedly – 6 times on average – through the reactor until the fuel has reached the specified burnup. More than 40 000 fuel pebbles have been tested in the AVR test reactor and irradiated to burn-ups of 10% fima. Behaviour under reactor operating conditions was found to be excellent and no fuel element damage was observed [79].
- As fuel compacts for use in HTGRs where the microspheres are embedded in a graphite matrix and where these fuel compacts are stacked in fuel rods. For the GT-MHR, TRISO-coated particles are bonded with a graphitic matrix to form cylindrical fuel compacts approximately 13 mm in diameter and 51 mm long. Approximately 3 000 fuel compacts are loaded into a hexagonal graphite fuel element, 793 mm long by 360 mm across flats, the same type of fuel element which showed excellent performance at the Fort Saint Vrain reactor. One hundred and two columns of the hexagonal fuel elements are stacked 10 elements high to form an annular core. Reflector graphite blocks are provided inside and outside of the core.

Figure 4.1 **TRISO coated fuel particles**



Early designs of HTGR-fuel used highly-enriched uranium (93%  $^{235}\text{U}$ ), and there was interest to reprocess these fuel elements for recovery of valuable fuel materials. Today only once-through irradiation of the low-enriched uranium fuel is envisaged as very high burn-ups are achieved which would make reprocessing technologically difficult and costly. Today, interest has also arisen to use such HTGR-fuel for the elimination of excess ex-weapons plutonium in HTR reactors.

#### 4.2.3.4 Status of technology in fuel design and fabrication

Table 4.2 gives an overview of the technological maturity of the fuel concepts discussed above.

Table 4.2 Overview of technological maturity of fuel fabrication

| Fuel type     | Technological Maturity |
|---------------|------------------------|
| UOX           | IV                     |
| MOX           | IV                     |
| IMF           | I                      |
| CERCER/CERMET | I                      |
| ROX           | I                      |
| Nitride       | I                      |
| Metal         | IV                     |
| HTGR          | III                    |

#### 4.2.4 Developments in reprocessing technology

Main developments:

- Operational improvements aiming at reduction of operational costs and releases to the environment.
- Separation technology for advanced fuel cycles (P&T) has known important developments and can separate the Transuranium and fission products with >99% recovery yields.
- However, target values for recovery yields should be specified and this needs an integrated view on advanced fuel cycle development.

Several national institutes currently are performing R&D on improved and new reprocessing technologies. Most of these technologies are linked to the partitioning and transmutation of actinides in advanced fuel cycles and can be categorised according to two schemes:

- Aqueous reprocessing techniques as an extension of the PUREX process in order to further separate the minor actinides and some of the fission products.
- Dry reprocessing techniques (in relation to oxide, nitride and metal fuels for FR/ADS) where the transuranics are separated from uranium and where, in principle, no further separation of the transuranics is performed.

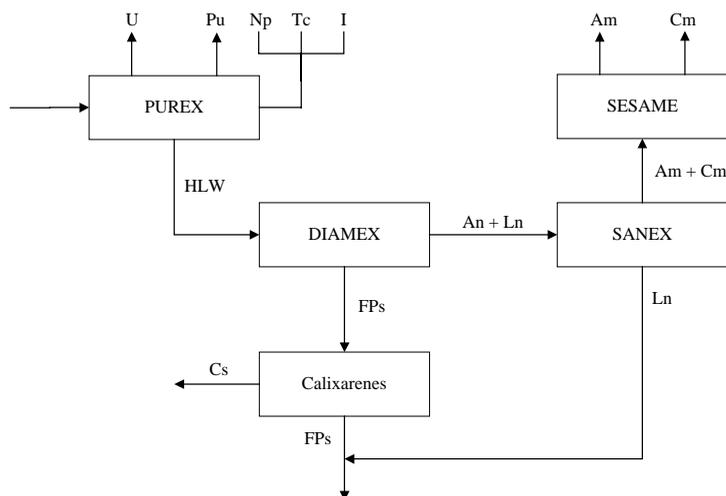
##### 4.2.4.1 Aqueous reprocessing techniques

The reference aqueous separation technology is the PUREX process as industrially developed in France and UK. This process is based on the use of tributyl phosphate (TBP), a complexant containing phosphorus and organic solvents, to separate actinides from an acidic aqueous medium. The basic principle of the PUREX process is that uranium and plutonium, whose stable oxidation states in nitric medium are VI and IV, respectively, are co-extracted by TBP and thus separated from the bulk of the fission products which remain in the aqueous phase. Uranium and plutonium are recovered with an industrial yield close to 99.9% (including losses in secondary wastes). Among the minor actinides, americium and curium are not extracted by TBP and remain in the aqueous phase. They accordingly follow the path of the fission products and currently are managed with the latter by conditioning in a

glass matrix. Neptunium is extracted by TBP, follows the uranium stream, and is separated from the latter in the second uranium purification cycle of the PUREX process. The effluent containing neptunium currently is added to the high-level waste stream. Hence, all the neptunium is sent to vitrification except for the portion distributed to the Pu product. Specific conditions are researched in JNC and CEA to co-extract Np quantitatively into product streams in the PUREX process. It is possible to use PUREX for reprocessing of FR-MOX fuel, and JNC has continued the development of it. The main differences of FR-fuel relative to LWR-fuel are higher Pu content, higher burn-up (which leads to higher FP content and activity and this accelerates the solvent degradation), geometric shape of assembly and fuel pin. To resolve these problems, JNC has conducted a system design study and developed equipment – such as rapid contactor, continuous dissolver, laser dismantling equipment, remote handling machine – relating to rapid processing, critical safety, solvent degradation, removing wrapper tube, etc. These technologies can be applied for LWR-MOX fuel even in case of multiple recycling. JAERI has studied and tested a simplified PUREX flow sheet for the separation of Np and Tc from the U-Pu stream [80].

The separation of the two other minor actinides, americium and curium, requires specific processes, as their oxidation state (III) in nitric acid is the same as that for lanthanide fission products, and they are therefore co-extracted with these fission products. Specific processes, i.e. TALSPEAK, DIDPA, TRUEX, DIAMEX and TPTZ, have been developed in order to separate these minor actinides from the fission products and subsequently from the lanthanides, with further separation of curium from americium by the SESAME process. The SETFICS process, a variation of the TRUEX process utilising DTPA as a selective strip reagent, was developed by JNC for the recovery of Am and Cm and separation of the actinides (An) from the lanthanides (Ln) (see Figure 4.2). More recently, a new family of extractants, BTP (Bis-Triazinyl-Pyridine), has been tested to separate the minor actinides from the lanthanides enabling a one-cycle SANEX process. This new extractant has proven to be very efficient for selective extraction of MA(III) at high acidity and shows good capabilities in centrifugal extractors. Tests at ITU have been performed in 2000 and indicated a recovery yield of 99.1% and 97.5% for Am and Cm, respectively [81]. These results represent an important break-through in the difficult field of minor actinide partitioning.

Figure 4.2 **Advanced aqueous reprocessing options starting from the PUREX process**



A thorough review of the state-of-the-art and developments in aqueous reprocessing technology has been given in literature [75] and the technological maturity of these processes is summarised in table 4.3.

Other wet separation processes under study are the “Four-group separation process” in Japan (based on the use of DIDPA) and the UREX-process in the US for separation of uranium from the transuranics with subsequent dry reprocessing of the fast reactor fuel.

Today’s reprocessing plants are continuously improving their economics and especially are reducing their possible environmental impacts. In Europe, an increasing pressure on reduction of radioactive discharges into the North Sea has led both Cogéma and BNFL to optimise their processes to reduce this environmental impact.

Table 4.3 Status of R&D on aqueous separation techniques [updated from 75]

|  | I  | II                       | III                      | IV                       | Remarks   |
|--|--|--------------------------|--------------------------|--------------------------|---|
| <i>U and Pu separation</i>   |  |                          |                          | <input type="checkbox"/> | achieved industrially by PUREX  |
| <i>Np separation</i>   |  | <input type="checkbox"/> | <input type="checkbox"/> |                          | 95% separation in PUREX<br>>95% separation by combination of PUREX, DIDPA, HDEHP, TRUEX |
| <i>Am + Cm separation:</i><br>• based on An/Ln co-extraction<br><br>• based on An selective extraction<br>• based on precipitation | <input type="checkbox"/><br><input type="checkbox"/> | <input type="checkbox"/> | <input type="checkbox"/> |                          | TALSPEAK<br>DIDPA, TRUEX, TRPO, DIAMEX<br>TPTZ, Picolinamides, CYANEX 301               |
| <i>Am separation in the oxidised state</i>   |  | <input type="checkbox"/> |                          |                          | Am/Cm separation using SESAME, BTP  |
| <i>Tc separation</i>   | <input type="checkbox"/>                             |                          | <input type="checkbox"/> |                          | soluble Tc by PUREX<br>insoluble Tc by PUREX  |
| <i>Tc – PGM separation</i>   |  | <input type="checkbox"/> |                          |                          |   |
| <i>I separation</i>  |  |                          | <input type="checkbox"/> |                          | 95% separation by PUREX   |
| <i>Zr separation</i>   |  | <input type="checkbox"/> |                          |                          | PUREX   |
| <i>Cs separation</i>   |  | <input type="checkbox"/> |                          |                          | Calixarenes or Zeolites   |
| <i>Sr separation</i>   |  | <input type="checkbox"/> |                          |                          | Titanic acid  |
| <i>Cs and Sr separation</i>  |  |                          | <input type="checkbox"/> |                          | Dicarbollides   |
| <i>Pd (PGM), Se, Ru separation</i>   | <input type="checkbox"/>                             |                          |                          |                          | Electrolytic extraction of soluble Pd, Se, etc  |
| <b>Other Processes</b>   |  |                          |                          |                          |   |
| <i>Four Group Separation Process</i>   |  | <input type="checkbox"/> |                          |                          |   |
| <i>UREX</i>  |  |                          | <input type="checkbox"/> |                          |   |

#### 4.2.4.2 Dry reprocessing techniques

The non-aqueous methods, or so-called dry or pyrochemical processes, have been developed in the USA, Japan and Russia in relation to their fast reactor programmes. The Research Institute of Atomic Reactors (RIAR) in Russia has been developing a pyrochemical reprocessing method, called DOVITA, based on the electrorefining of oxides combined with vibro-pack fuel fabrication since the early 1960s. The Central Research Institute of Electric Power Industry (CRIEPI) in Japan has been investigating the technology of recycling actinides from metallic FBR or reduced oxide fuels, and recovering them from high-level waste. The three processes make use of pyrochemical techniques in molten chlorides (LiCl-KCl) in contact with liquid metal (Cd or Bi). This process is based on a metal fuel cycle previously developed at the Argonne National Laboratory (ANL) in the USA. The advantage is that it can be applied to either oxide or metal fuel, of any composition whether uranium only or mixed with plutonium, and to fuels of very high burn-up. Expected advantages are improved proliferation resistance, simplified criticality control and economically compact equipment. The disadvantage, however, is the lack of industrial experience with a high-temperature process using corrosive materials.

In September 2000, the United States began electrometallurgical refining to treat sodium-bonded fuel from the Experimental Breeder Reactor-II (EBR-II). Development of this technology concluded in 1999 after the successful demonstration of the process. Operations to treat the fuel that contains about 25 tons of heavy metal will continue through 2010.

New technologies and processes based on chloride volatilisation are introduced as alternatives, and important progress has been reported, especially in the USA [82]. The fluoride volatility method was investigated in US, Russia, France, Japan, etc. in the 1950s-80s. It has the characteristic of being a simple process and Russia has been continuing its development for reprocessing spent fuels.

Table 4.4 summarises the main differences and maturity of the dry processes compared to wet processes [83].

**Table 4.4 Characteristics of reprocessing with aqueous or dry processes**

| <b>Process</b>   | <b>Aqueous process</b>  | <b>Dry process (metal-based)</b>  | <b>Dry process (oxide-based)</b>   |
|--|---|---|--|
| Feed of material   | Continuous flow   | Batch   | Batch  |
| Solvent to be used                                       | Organic solvent, nitric acid  | LiCl-KCl, Cd, Bi  | NaCl-KCl-CsCl  |
| Operating temperature                                    | Below 100°C   | Mostly 500°C, partly >1 000°C   | 630-650°C, partly >1 000°C   |
| Fuel type to be applied                                  | Oxides  | Metal, oxides, nitride  | Oxides   |
| Applicability for Pu recycled fuel or short cooling fuel | Need to maintain a quite short contact time between nitric acid and solvent | No special considerations needed  | No special considerations needed   |
| Product form and purity                                  | Oxide highly pure product   | Metal: U, mixture of TRU with some amount of rare earth   | Oxide: UO <sub>2</sub> +NpO <sub>2</sub> with Ru, Rh, PuO <sub>2</sub> with some amount of americium oxide |
| Recovery ratio for Pu, %                                 | >99.9   | >99.5 (to be demonstrated)  | >99.3-99.7   |
| Minor actinides recovery                                 | Need additional process, such as TRUEX                                      | No need additional process  | New method has to be developed, such as, DOVITA  |
| Fuel fabrication   | Pelletising, Vibro-packing  | Injection casting   | Vibro-packing  |
| Criticality  | Need severe control   | Need relatively mild control  | Need relatively mild control   |
| HLW form before solidification                           | Oxide   | Chloride  | Phosphate and chloride   |
| Secondary waste  | Degraded solvent  | Chloride salt (LiCl-KCl base)   | Chloride salt (NaCl-KCl-CsCl base)   |
| Waste treatment and solidification                       | Solidification into boro-silicate glass                                     | Direct solidification into artificial rock (SODALITE), or solidification into boro-silicate glass after waste treatment | Solidification into phosphate glass  |
| Material for apparatus                                   | Stainless steel, Zr-lined steel   | Steel, need to develop long-life material for >1 000°C processes  | Pyrographite, need to develop long-life material for high temperature processes                            |
| Accountancy  | Continuous accounting is possible   | Need batch-wise accounting measure  | Need batch-wise accounting measure   |
| Economics  | Based on experience   | Under estimation  | Under estimation   |
| Technological maturity                                   | Highly matured  | Under demonstration by laboratory and pilot scale facility  | Under demonstration by laboratory and pilot-scale facility   |

### 4.3 Developments in reactor technology

Main developments:

- Evolutionary reactor developments exist but lack market potential.
- New designs are emerging but need international development framework.

While reactor technology is not part of this study, developments in reactor technology can affect the trends in fuel cycle developments. Some reactor developments will, therefore, be described briefly in order to introduce their role and relative merits/drawbacks within their fuel cycle scheme. This section will briefly describe five reactor concepts that are considered relevant in relation to the fuel cycle schemes discussed in Chapter 5:

- Advanced light water reactors (ALWR).
- High-temperature gas-cooled reactors (HTGCR).
- Fast reactors (FR).
- Molten salt reactors (MSR).
- Accelerator-driven systems (ADS).

#### 4.3.1 Reactor development in the short to medium term

##### 4.3.1.1 Advanced light water reactors

Evolutionary designs of today's light water reactors have been proposed in the past years as third generation type of plants. Several designs are currently proposed by the nuclear industry, e.g. EPR by NPI (Siemens/Framatome),<sup>57</sup> System 80+ and BWR90+ by ABB, APWR/ABWR in Japan,<sup>58</sup> KNGR based on the ABB's System 80+ design, CANDU 6 by AECL. These third generation LWRs incorporate higher safety standards by use of passive safety features,<sup>59</sup> improved economics, and are based on feedback of operating experience with current LWRs. Detailed descriptions of these designs have been given in literature [84] and will not be presented in this report. These designs should be viable systems for the short to medium-term market conditions, while even more advanced "revolutionary" systems could be envisaged only for the longer term (see following sections).

The impact on the nuclear fuel cycle by these evolutionary designs is limited essentially to the amount and type of fuel assemblies used. These designs do not make use of new types of fuel, and only envisage higher average burn-ups and/or enhanced MOX loading of the current fuel designs. The

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57. The EPR envisages a preventive maintenance concept leading to an improved average plant availability of 92% during the 60 years lifetime of the plant. In total 241 fuel assemblies (for a 4 900 MWth design) would be loaded in the reactor core with an average fuel burn-up of 60 GWd/tHM in order to reduce the fuel cycle costs.

58. The Japanese APWR core consists of 257 fuel assemblies of an advanced 17x17 type and incorporates flexibility to meet future requirements such as operation with MOX fuel in more than 1/3 of the core and fuel burn-up of more than 55 GWd/tHM.

59. All of these designs claim a lower core damage frequency, in the range of  $10^{-5}$  down to less than  $10^{-6}$  for the cumulative frequency of large releases following a core damage. These core damage frequencies are set forward in the utility requirements for future advanced reactor concepts.

evolutionary designs of fuel assemblies, as mentioned in Chapter 2 and preceding sections in this chapter, are used in the reactors without any major modification.

While today's socio-political constraints to nuclear indicate a specific concern related to the back-end of the fuel cycle, i.e. waste management and the long term radiological risks, the reduction of mass and radioactivity of plutonium and some minor actinides by transmutation in such LWRs encounters specific problems. It should be remarked that, in essence, a LWR type of reactor does not efficiently use the energy potential contained in the mined uranium and results in a high ratio of radioactive waste versus energy produced. In addition, in order to cope with the transmutation or burning of the minor actinides a fast reactor spectrum would be needed. Further, if one would consider using a (homogeneous or heterogeneous) multiple recycling strategy of the (Pu+MA)-fuel in LWRs, one can conclude that all mass flow rates in the fuel cycle would be increased, (e.g. the curium quantities are three times higher). Another effect is related to the presence of lanthanides in the fuel, which affect the reactivity balance in a thermal reactor. Fabrication of the needed targets (homogeneous or heterogeneous) is not feasible in current fuel fabrication facilities and therefore dedicated facilities, favouring the heterogeneous recycling mode, should be envisaged.

#### *4.3.1.2 High temperature gas-cooled reactors*

A revised interest in this reactor type has been triggered during the past years, mainly due to considerations on safety, energy efficiency and fuel cycle. The high temperature gas cooled reactor (HTGR) is an advanced nuclear energy technology employing a ceramic coated-particle fuel (see Section 4.2.3.3), a graphite moderator (which also serves to establish core geometry) and a single-phase, inert gas (helium) coolant. This particular combination of fuel, moderator/core structure and coolant provides a capability for high temperature operation that is rather unique among current nuclear reactor concepts. The HTGR has been under development since the 1960s and 5 helium-cooled reactors, which operated in the 1960s, 70s and 80s demonstrated the inherent technological characteristics of helium cooled reactors. In the 1980s, modular helium reactor (MHR) designs were developed, in both Germany and the USA, that demonstrated inherent passive characteristics for meeting stringent safety goals without relying on active safety systems or operator action. Unfortunately, a major drawback of these passively safe designs, which employed the Rankine (steam turbine) power conversion cycle, was that their economic performance was, at that time, not competitive with LWRs. However, by the early 1990s, advances in industrial gas turbine (GT) technology, highly effective recuperators and related equipment, allowed the coupling of the passively safe capability of a modular helium reactor (MHR) with an efficient Brayton cycle (gas turbine) to produce sufficient economic potential to reignite interest in the design.

In March 1991, JAERI started construction of the HTTR (High Temperature Engineering Test Reactor) [85], which is a high-temperature gas-cooled reactor employing the TRISO coated-particle type fuel system (see Figure 4.1) [86], and its first criticality was attained in November 1998. The characteristics of this reactor are presented in Table 4.5. The reactor outlet coolant temperature of 950°C is the highest achieved in any reactor. The HTTR is utilised for establishing and upgrading the technology bases for advanced HTRs, including nuclear heat application such as hydrogen production [87], and for various innovative high temperature basic studies. A feasibility study of an HTGR-GT system was also carried out recently [88].

In 1995, the construction of a pebble-bed type high-temperature gas-cooled reactor (HTR-10) started in China, with a thermal power output of 10 MW. The first criticality has been reached in December 2000.

Table 4.5 **HTTR specification** [85]

|                             |  |
|-----------------------------|--|
| Thermal power               | 30 MWth  |
| Outlet coolant temperature  | 850°C rated operation<br>950°C high-temperature test operation |
| Inlet coolant temperature   | 395°C  |
| Fuel                        | Low-enriched UO <sub>2</sub>                                   |
| Fuel element type           | Prismatic block  |
| Direction of coolant-flow   | Downward   |
| Pressure vessel             | Steel  |
| Number of main cooling loop | 1  |
| Heat removal                | Intermediate heat exchanger pressurised water cooler           |
| Primary coolant pressure    | 4 MPa  |
| Containment type            | Steel  |
| Plant lifetime              | 20 years   |

In the modular high-temperature reactor, the high temperature capabilities have been optimised to achieve increased efficiency, combined with simplified safety systems and enhanced safety margins. Specifically, the reference Brayton cycle (gas turbine) cycle for electricity generation would achieve an overall efficiency approaching 48%. The GT-MHR project by General Atomics and affiliates (see table 4.6), couples an MHR, contained in one vessel, with a high efficiency gas turbine energy conversion system contained in an adjacent vessel. The reactor and power conversion vessels are interconnected with a short cross-vessel and are located below ground level in a cylindrical silo. The reactor employs a graphite moderator and TRISO coated particle fuel. Today, the GT-MHR conceptual design work is being directed by a joint steering committee composed of representatives from General Atomic, Minatom, Framatome and Fuji Electric. Table 4.7 presents a comparison of the GT-MHR with a conventional LWR, showing some principal fuel cycle characteristics as well as some parameters (e.g. thermal discharge, spent fuel arising) relevant to environmental impact comparisons. Based on a co-operative agreement in 1993, General Atomic and Minatom envisage applying GT-MHR technology to dispose of weapons-grade plutonium. The GT-MHR plant design and development and construction would cost about USD 453 million, and the fuel design and development, together with the construction of a fuel fabrication facility, would add USD 115 million.

Table 4.6 **Technical characteristics of the GT-MHR**

|                                   |               |
|-----------------------------------|---------------|
| Reactor power                     | 600 MWth      |
| Core Inlet/Outlet temperatures    | 490/850°C     |
| Core Inlet/Outlet pressures       | 7.07/7.02 MPa |
| Helium mass flow rate             | 320 kg/s      |
| Turbine Inlet/Outlet temperatures | 848/511°C     |
| Turbine Inlet/Outlet pressures    | 7.01/2.64 MPa |
| Net electrical output             | 288 MWe       |
| Net plant efficiency              | 48%           |

Table 4.7 **Principal characteristics GT-MHR compared to LWRs**

| Reactor                                   | LWR  | GT-MHR |                |
|---|------|--------|----------------|
| Unit thermal power                        | 3914 | 600    | MWth           |
| Power units/plant                         | 1    | 4      |                |
| Plant thermal power                       | 3914 | 2 400  | MWth           |
| Thermal efficiency                        | 35   | 48     | %              |
| Plant electrical power                    | 1385 | 1 152  | MWe            |
| Thermal discharge*                        | 1.8  | 1.1    | GWth/GWe       |
| Equilibrium fuel cycle                    |      |        |                |
| Heavy Metal Loading                       | 26.8 | 7.5    | MT/GWth        |
| Enrichment                                | 4.2  | 15.5   | %              |
| SWU                                       | 135  | 221    | kgSWU/GWe-year |
| Burn-up                                   | 47.8 | 121    | GWd/tHM        |
| U <sub>3</sub> O <sub>8</sub> Consumption | 181  | 246    | MT/GWe-year    |
| Spent fuel                                |      |        |                |
| Discharged HM                             | 21.4 | 5.4    | MT/GWe-year    |
| Enrichment                                | 0.9  | 4.8    | %              |
| Discharged Pu                             | 235  | 109    | kg/GWe-year    |
| Discharged <sup>239</sup> Pu              | 171  | 43     | kg/GWe-year    |

\* Thermal discharge = 1/(Efficiency/100) -1

Another HTGR project, the pebble-bed modular reactor (PBMR), led by the South-African company, ESKOM, has received interest within some OECD Member countries. The South African government gave the go-ahead in April 2000 to the PBMR Project to perform a detailed feasibility study as well as to carry out the EIA process, which allows for public participation. The EIA and detailed feasibility study for the demonstration module will be inputs to the decision by ESKOM on whether to seek Government approval to move to the next phase. This decision is not expected to be taken before the middle of 2001. Although the final choice of the location of the demonstration plant would be known only once the EIA has been concluded, the currently preferred site is Koeberg, which is the site of ESKOM's currently operating LWRs, owing to the already existing infrastructure, experience and qualified personnel. Subject to Government approval, the construction of a demonstration plant is scheduled to start mid-2001, with commercial operation projected for 2005. A half-scale model of the reactor went critical at the Kurchatov Institute in Moscow in July 1999.

#### *Fuel cycle aspects of HTRs*

Some unique problems exist for the storage, transportation and disposal of spent fuel from the GT-MHR. The volumes of fuel to be disposed of are relatively large when the complete prismatic blocks containing the fuel are disposed directly. Some early work in the USA indicated that the lifetime of the HTGR graphite in a repository was several orders of magnitude longer than for LWR spent fuel. As a consequence, it was concluded that it would be desirable to dispose of the entire block in order to minimise radionuclide releases. If the fuel pellets are removed from the prismatic blocks for separate disposal, specialised techniques will be required to limit the release of radioactive materials (e.g. <sup>14</sup>C, <sup>110m</sup>Ag). Even if the graphite prismatic blocks would be recycled (after removal of the particles), they will have to be disposed of eventually, since the number of fuel cycles for a prismatic block is limited to 3 or 4 by physical changes in the graphite during irradiation. The relatively high cost of reactor grade graphite (currently estimated to be in the range of USD 75/kg) may eventually justify reprocessing of the graphite for refabrication into new blocks.

In Germany, the AVR pebble-bed HTR spent fuel has been stored for an interim period in vault-type storage facilities consisting of stainless steel racks with 72 storage canisters each, with each

canister containing about 1 000 spherical fuel elements (fuel balls). The total activity of stored fuel amounts to about 600 000 Curies, including about 10 000 Curies of  $^{85}\text{Kr}$  and 2 000 Curies of tritium. No release of radioactivity has been detected to date and the maximum surface temperature of the sealed canisters has been only 10°C above the ambient air inlet temperature. This and other work suggest that the repository performance of the fuel will be better than that of LWR spent nuclear fuel, because of the chemical inertness of nuclear-grade graphite. The volume of spent nuclear fuel per unit of electricity produced in HTRs is similar to that for the LWR. The graphite in the spent fuel increases the waste volume, but this is offset by the higher fuel burn-up and higher efficiency of the HTR.

### *Economics of MHTGRs*

A study by ECN in the Netherlands on the economic viability of MHTGRs compared to LWRs indicated that the total investment cost for a 1 000 MWe LWR plant and for a n<sup>th</sup>-of-a-kind 4×200 MWe MHTGR plant would be comparable, but with somewhat higher specific investment costs (expressed in USD/kWe ) for the MHTGR [89]. The operation and maintenance staff levels as well as the fuel cycle costs were estimated to be quite comparable, i.e. 0.45 person/MWe and USD28/kWe-year, respectively, for LWRs and 0.45 person/MWe also and USD22/kWe-year, respectively, for MHTGRs. This analysis concluded that the estimated electricity generation cost for LWRs and MHTGRs (in the year 2010) would not differ significantly.

### **4.3.2 DUPIC fuel cycle**

Based on the neutron economy of heavy water, CANDU (PHWR) reactors can burn a variety of fuels, including natural uranium and slightly enriched uranium, and even has the potential to reuse the spent fuel from LWRs. Today's CANDUs use natural uranium as fuel, with the plutonium that is produced during irradiation contributing about 50-70% of the energy production at the end of an irradiation cycle. The large contribution of Pu to the energy production at the end of fuel cycle points to the possible benefit of extending the fuel life by the use of slightly enriched uranium. With an enrichment level of 1.2%  $^{235}\text{U}$ , the fuel burn-up would be extended from 7 GWd/tHM to 22 GWd/tHM, while the natural uranium consumption would be reduced from 157 tonnes/GWe-year to 114 tonnes/GWe-year.

As spent LWR fuel with a burn-up of 35 GWd/tHM still has a fissile content of some 1.5% (about 0.9%  $^{235}\text{U}$  and 0.6% fissile Pu), this spent fuel, after removal of the fission products, could be reused in a CANDU reactor to obtain an additional energy output of some 27 GWd/tHM. Such a fuel cycle, referred to as "Direct Use of spent PWR fuel in CANDU", or simply "DUPIC" [90], is described in Section 5.3.2.

### **4.3.3 Reactor developments in the medium to long term**

#### *4.3.3.1 Fast reactors*

The liquid metal cooled fast reactor (LMFR) technology has been developed in many countries, including France, Germany, India, Japan, the Russian Federation, the United Kingdom and the USA. The current development of fast reactor technology is undergoing difficulty, particularly in the context of an overall slowing-down of nuclear power development. Some governments have drastically cut back the R&D budget for fast reactors, and others are considering shelving the present breeder technology altogether. However, LMFR technology, when fully developed, is believed to be the best technology capable of utilising depleted uranium, transforming long lived transuranic waste and enabling the efficient use of both military and civilian plutonium, low grade uranium and thorium. In addition to using fissile resources more efficiently, FRs also have potential for use as "waste burners"

in advanced nuclear fuel cycles incorporating partitioning and transmutation of fission product isotopes. There is also renewed and increasing interest in gas-cooled fast reactors, since in addition to the possibility to offer higher thermodynamic efficiencies, they may exhibit advantages with respect to fuel, safety and operability.

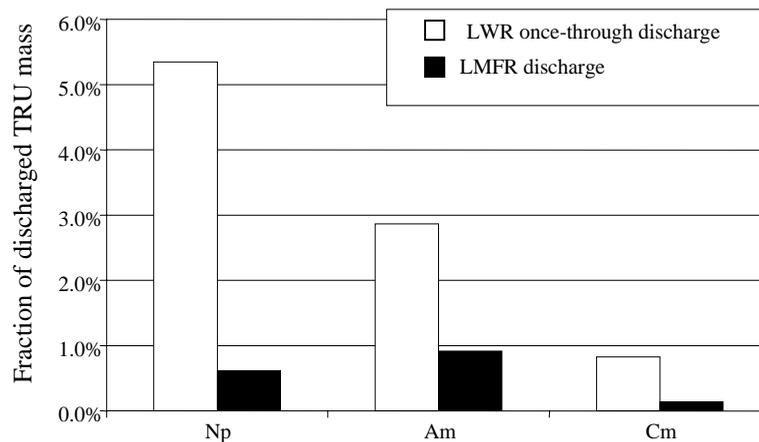
Fast reactors can be designed as transuranic (TRU) burners aiming for a reduction of the TRUs leaving a LWR reactor park. As the fast TRU-burner currently only may achieve a TRU-reduction of about 14% in a single irradiation, about 15 recycle loops (i.e. around 100 years elapsed time) are needed to achieve a 90% destruction of the TRUs.

Japan plans to carry forward steadily and systematically the development of FBRs. The experimental fast reactor Joyo (100 MWth) has been in operation since 1977, and has yielded various irradiation test and operation data. In order to improve its performance for irradiation testing, it is under-going core modification to increase its power level (to 140 MWth), to achieve higher neutron flux and to extend test capabilities. The prototype FBR Monju (280 MWe) is under pre-operation test to master the power plant technology, improve performance and establish economic efficiency. It reached first criticality in April 1994 and started generating of electricity in August 1995, but a sodium leak occurred in the secondary cooling system in December 1995. The investigations of the cause of the accident and total safety evaluation of the Monju plant have been completed. Monju will be a demonstration of FR technology with regard to sodium handling, core management, power generation system, maintenance, etc. The reactor is also an experimental facility for irradiation tests and other innovative tests. JNC initiated a feasibility study to pursue the economic improvement, reduction of environmental impacts and other advantages of FBR and related fuel cycle in collaboration with electric power companies and other related organisations in 1999. In the study, various options are evaluated and conceptual designs are examined, considering the attainable perspectives for: ensuring safety, economic competitiveness, efficient utilisation of resources, reduction of environmental burden, and enhancement of non-proliferation. Reactor design studies have been carried out by combining various coolants (sodium, heavy-metal, gas, etc.) and fuel types (MOX, nitride, metal, etc.). The results will be evaluated by multi-criteria analysis and the feasible candidate concepts for FRs, reprocessing systems and fuel fabrication systems will be selected in Phase-1. After conducting key experiments and detailed design studies, some of the most promising concepts will be chosen in Phase-2. The conceptual design study and selection of the most promising FR technologies and fuel cycle systems would be completed by about 2015.

While FR discussions in this study have been limited to MOX fuelled types, other concepts are under study as well. Metal-fuel fast breeder reactors, for example, show performance suitable for the consumption of minor actinides, while the associated pyrochemistry is compatible with the recycling of minor actinides. In this case, the minor actinide (MA) content in the feed is limited to a few weight percent, in order to keep the core safety parameters acceptable and to facilitate a simple casting method of fuel fabrication. The results of studies show that the plutonium and minor actinides from LWRs can be consumed without further accumulation by establishing a complementary FR power generation reactor with its fuel cycle. Other developments involve the use of molten salt type reactors which, when coupled to pyrochemical separation, could integrate the separation and transmutation operations and achieve higher net transmutation rates than other systems.

Figure 4.3 summarises the transuranic composition for a LWR and a typical LMFR. A self-sufficient LMFR using the recycled transuranics from LWR discharge as its start-up fissile source would, in addition to producing power over its operating life, result in an equilibrium fuel composition with reduced long-term radiological toxicity owing to the reduced proportion of higher actinides. Of particular importance is the reduced fraction of  $^{241}\text{Pu}$ , whose decay product ( $^{237}\text{Np}$ ) dominates the long-term hazard.

Figure 4.3 Comparison of LWR and LMFR transuranic composition [91]  
(Pu amounts to 91% of TRU for LWR discharge and 89% for LMFR)



The use of dedicated MA fast burners with important MA loads (ideally 100% of the reactor fuel), would allow the reduction of specific fuel-cycle operations (fabrication, handling, storage and reprocessing) and limit safety countermeasures to a small fraction of the reactor fleet with respect to the homogeneous option. Those dedicated burner reactors have to take into account degraded core characteristics (e.g. smaller Doppler and void reactivity coefficients) due to the increase of the MA-fraction. Therefore, the use of small cores, increasing the leakage effects, and the use of “cold” fuel, e.g. nitride fuel, are main axes of R&D if one considers the use of solid fuel type burner reactors. Based on the lower reactivity losses, and as such, possibly longer fuel residence times in combination with multiple recycling, this scheme could well increase the transmutation potential.

It might therefore be sensible to consider a scenario in which fast reactors are, during their initial introduction, used in a once-through cycle, in which case they can be used most effectively to reduce the quantity of LWR-TRUs in storage. In the longer term, as more fast reactors enter into operation, the fast reactors change to a recycle function. This would then effect a long-term reduction in radio-toxicity. The main objective in fast reactor burner development is therefore the achievement of maximum destruction rates of TRUs in a cycle. The ADSs are thought to be better for these functions based on their more flexible fuel management and the harder neutron spectra achievable. A comparative study is, however, necessary to support this perceived conclusion.

#### 4.3.3.2 Molten salt reactors

In the last several years, there has been a renewed interest in molten salt reactors (MSRs). During the 1960s, the United States developed the Molten Salt Breeder Reactor (MSBR) as the primary backup option for the liquid-metal fast breeder reactor. The United States had a large technology development programme that constructed and demonstrated the MSR technology using an 8 MWth reactor. That reactor development programme, an outgrowth of an earlier Aircraft Nuclear Propulsion (ANP) programme, was terminated in the early 1970s.

In a MSR, the fuel is a liquid mixture of lithium and beryllium fluoride salts with thorium and <sup>233</sup>U fluorides dissolved in the salt. The reactor core consists of unclad graphite stringers with holes through which the salt flows. The fuel salt (700°C exit temperature) in the low-pressure primary system flows to external heat exchangers that transfer the heat to a secondary fluid and then flows

back to the reactor (570°C inlet temperature). The secondary fluid is sent to a steam generator to produce steam for production of electricity.

The MSR normally would operate on a  $^{233}\text{U}$ -thorium fuel cycle after start-up on low-enriched uranium. The fission products produced during operations dissolve into the salt. A small secondary salt-processing loop removes the fission products from the molten salt and solidifies the waste for disposal in a repository. Actinides remain in the reactor until they fission or are converted to higher actinides, that ultimately fission.

There has been renewed interest for the Molten Salt Reactor in Japan, Russia, France, and the United States based on several considerations:

- *Waste Management.* A molten salt reactor has a radically different fuel cycle than solid fuel reactors. The reactor normally destroys long-lived actinides rather than sending actinides to the waste streams. This changes the characteristics of the wastes destined for a repository resulting in waste forms with greatly reduced decay times.
- *Non-proliferation.* The inventory of weapons-usable fissile materials in a MSR is much smaller than for other reactor concepts. The plutonium isotopic composition is very unusual, with  $^{242}\text{Pu}$  being the dominate isotope, and it would be quite difficult to build a weapon with such plutonium. The primary fissile material ( $^{233}\text{U}$ ) in the reactor can be made non-weapons usable by addition of  $^{238}\text{U}$  to the fuel salt.
- *Safety.* The safety system is unique. In an emergency, the fuel is drained from the reactor to passively-cooled storage tanks. This may imply that passive cooling can be assured for any size reactor, whereas most reactor concepts can assure passive emergency core cooling only up to certain maximum reactor sizes.
- *Technology.* The early research and tests showed the technical viability of MSRs, although there were significant difficulties. Technology advances have provided potential solutions to address many of the difficulties associated with earlier MSRs.

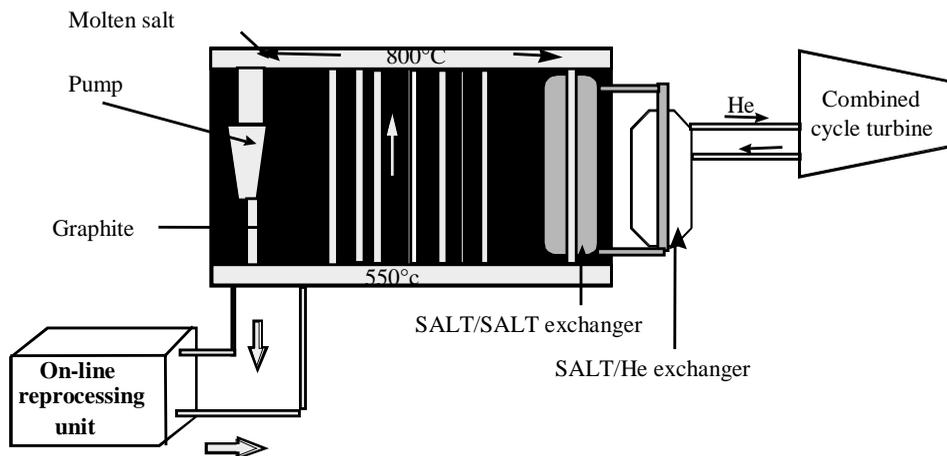
In France, the electricity utility EdF is currently studying a concept for a graphite-moderated molten salt reactor, entitled AMSTER (see Figure 4.4) [92], in which the salt treatment system has been redesigned in order to reduce waste production. Using this concept, one can define a large number of configurations according to the products loaded and recycled.

A number of studies are planned and underway that should determine whether changing goals and improved technologies will make the MSR an attractive future reactor type. The fuel cycle options for the molten salt reactor are described in Section 5.4.3.

In the United States, several laboratories (e.g. ORNL and Sandia) are examining a modular, molten-salt-cooled, graphite-matrix-fuel, advanced high-temperature reactor (AHTR) to provide heat at the conditions (high temperatures and low pressures) necessary to create new nuclear energy options:

- Hydrogen production by thermochemical water splitting (requires 800 to 1 000°C heat).
- Advanced electric production methods (indirect gas turbine cycles and direct thermal-to-electric techniques).

Figure 4.4 Schematic diagram of the AMSTER molten salt reactor concept [92]



The AHTR would generate up to 600 MW(t) with outlet temperatures of  $>1\ 000^{\circ}\text{C}$  [93]. The fuel is a graphite-matrix fuel with the same general characteristics as that developed for modular high-temperature gas cooled reactors (MHTGRs). Such fuels have been demonstrated at temperatures up to  $1\ 200^{\circ}\text{C}$ . The AHTR fuel cycle would be similar to that for the MHTGR. The molten salt coolant is the same (but without fissile, fertile, or fission products) as proposed for conventional molten salt reactors. Molten salt is being considered as a coolant because of its high boiling point ( $\sim 1\ 400^{\circ}\text{C}$ ) and excellent heat transfer properties. The very high boiling point allows high-temperature operation at low pressure, which has safety and economic advantages when heat at high temperatures and low pressure is required. The liquid coolant has better heat transfer properties than helium gas. Consequently, the reactor may operate at somewhat higher coolant exit temperatures than an MHTGR, given the same fuel temperature constraints.

#### 4.3.3.3 Accelerator-driven systems (ADS)

In the past years, a number of proposals for applications of accelerator driven systems (ADS) have been presented, varying from an energy amplifier with lumped  $\text{Th}/^{233}\text{U}$  fuel and fast spectrum with lead coolant, to fission product and actinide transmutation with fast and thermal spectra. The main argument usually given for the ADS concepts is improved safety compared to critical reactor systems, but this argument is of secondary nature. The more decisive arguments for ADS rely on neutron economy and fuel flexibility in options for burning the minor actinides in advanced fuel cycle schemes. Most of the present design studies focus on four categories: sodium cooled solid fuel fast reactor technology; liquid metal cooled options with solid fuel favouring harder neutron spectra and reducing the effects of positive void reactivity coefficients and technological limits; gas-cooled solid fuel systems; and concepts relying on molten salt technologies.

The accelerator driven system concepts are most specifically designed for transmutation purposes as dedicated transmuters to be deployed in a partitioning and transmutation cycle. In fact, two main options emerge if one would opt to fully close the nuclear fuel cycle. The first option consists of using a fast reactor recycling not only the plutonium but also the minor actinides, e.g. the IFR-concept. The second option consists of concentrating the minor actinides, already being a small mass, in a separate part of the fuel cycle (second stratum) where these minor actinides are burnt in dedicated transmuters, i.e. accelerator-driven systems. The plutonium would be burnt in fast reactors in the main fuel cycle (first stratum).

The choice between options does not rely only on the choice of reactor system but also on the type of fuel and reprocessing chosen. Where the first option implies the use of pyrochemical reprocessing, both wet (for the first stratum) and dry reprocessing (for the second stratum) would need to be used in the second option. The second option would benefit from concentrating the highly-active materials (requiring remote handling) in a separate part of the fuel cycle where the first option would rather imply a more distributed handling of these materials. However, the use of pyrochemical reprocessing would relax these problems considerably. As the transmutation effectiveness of an accelerator driven system can be higher than for a critical fast reactor, an ADS-scheme could more rapidly adapt to fuel cycle changes, e.g. breeding or phase-out scenarios.

A very important and distinctive aspect of all ADS designs relates to the very demanding accelerator requirements. Not only are the beam energy and intensity rather high, but also the reliability of the beam, especially to avoid troublesome beam trips, is a key area for future developments (e.g. reduction by at least a factor of 100 in the frequency of beam trips should be obtained). Material and technological problems related to the beam window are very important issues, and future R&D should be focused on reducing the need for the overdesign. Other very important R&D needs cover the thermodynamic aspects of the ADS sub-critical cores, in particular for the spallation target, and aspects related to the material irradiation of target window and structural and fuel materials, the control and safety aspects, etc.

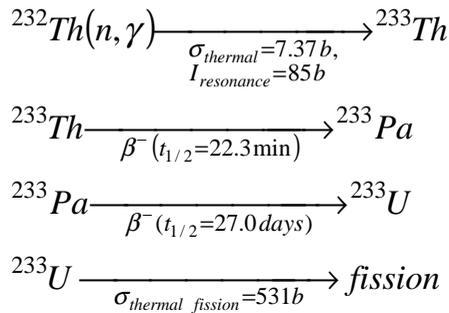
ADS concepts are developed mainly in advanced nuclear fuel cycles for direct burning of the TRUs from a LWR-park (i.e. ATW-project in USA) or with so called “Double Strata” fuel cycles (i.e. OMEGA in Japan, and GEDEON in France). These fuel cycles belong, however, to the more radical changes in fuel cycle options and are discussed in more detail in Chapter 5.

## 4.4 Other potential changes in the fuel cycle

### 4.4.1 Thorium fuel cycles

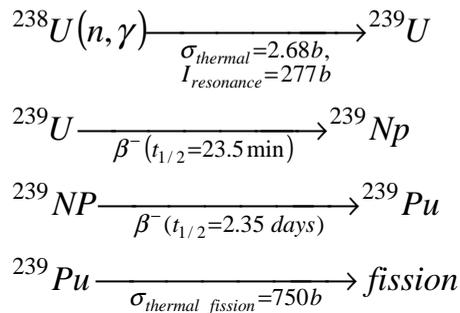
The idea of using thorium as a nuclear fuel, instead of uranium, is one of long standing. It is important to appreciate from the outset, however, that thorium is not a substitute for natural uranium, since none of the isotopes of thorium are fissile (i.e. capable of fission through the use of low-energy neutrons). The only naturally occurring fissile isotope is  $^{235}\text{U}$ , which constitutes 0.7% of natural uranium. The use of thorium-fuel in a once-through scenario would, therefore, require uranium or plutonium as a supporting source of fissile material. Reprocessing of a significant fraction of the spent fuel (in order to separate the fissile  $^{235}\text{U}$ ) would be a necessity if one would embark on a thorium closed fuel cycle. Therefore, the thorium fuel cycle should be considered as a long-term development, while a nearer term use of thorium could be in Transuranium burning systems.

Thorium is however a useful material in nuclear reactors because neutron capture in  $^{232}\text{Th}$  leads to  $^{233}\text{U}$ , a fissile isotope of uranium with nuclear properties to similar those of the naturally occurring  $^{235}\text{U}$ . The reactions and decays leading to  $^{233}\text{U}$  are shown in the following equations.



### Production of $^{233}\text{U}$ from $^{232}\text{Th}$

For comparison, the other common fertile isotope,  $^{238}\text{U}$  produces the fissile isotope  $^{239}\text{Pu}$  through the reactions and decays shown below.



### Production of $^{239}\text{Pu}$ from $^{238}\text{U}$

#### History of thorium fuels

It was noted at an early stage of nuclear technology development that  $^{233}\text{U}$  represents a superior fissile nuclide, in that it produces more neutrons per thermal neutron absorbed than any other fissile isotope. This feature, and the fact that thorium is much more abundant as a natural ore than uranium, prompted numerous attempts to design and implement a nuclear reactor based on thorium fuel.

In the USA, thorium dioxide was the primary component in the nuclear fuel used in the reactors at Elk River, Indian Point (No. 1), the Peach Bottom High-temperature gas-cooled reactor experiment (HTGRE) and in the Shippingport Light water breeder reactor (LWBR) [95]. In all these cases, the  $\text{ThO}_2$  fuel performed satisfactorily, showing exceptionally good dimensional stability and a fission gas release of only 1 to 2%. The fuel for the HTGRE consisted of two types of microspheres, fissile and fertile, randomly intermixed in a carbon matrix. The fissile microspheres contained highly enriched uranium carbide coated with a single layer of high-density pyrolytic carbon. These “BISO” coatings released unacceptable amounts of fission gases and the BISO fuel elements were used to only half of their design life. The second core contained “TRISO” microspheres. Fuel elements containing thorium microspheres of the TRISO design were tested to burn-ups as high as 15 at% of the initial thorium. Fission gas release was very low, only some 0.1% of the total fission gas produced.

The use of fuel containing both thorium and uranium reduces the total plutonium production per MWd by a factor of 3.2, compared with conventional  $\text{UO}_2$  fuel burned to 45 MWd/kg. Production of  $^{239}\text{Pu}$  per MWd is a factor of about 4 less in the  $\text{ThO}_2\text{-UO}_2$  fuel than in conventional fuel. The plutonium produced is high in  $^{238}\text{Pu}$ , leading to a decay heat about 3 times greater than that from plutonium derived from conventional fuel burned to 45 MWd/kg and 20 times greater than weapons grade plutonium. This decay heat makes handling and fabrication of any separated plutonium more difficult. Spontaneous neutron production from the plutonium in the  $\text{ThO}_2\text{-UO}_2$  fuel is about 50% greater than that from conventional fuel. The isotopic constituents of the plutonium are shown in Figure 4.5. For comparison, the plutonium composition in a conventional  $\text{UO}_2$  core is shown in Figure 4.6.

Thorium dioxide has a higher thermal conductivity than  $\text{UO}_2$ . Therefore  $\text{ThO}_2\text{-UO}_2$  fuel will operate cooler, and will retain within the fuel more of the fission products, especially the gases.  $\text{ThO}_2\text{-UO}_2$  fuel probably can be successfully operated to higher burn-ups than  $\text{UO}_2$  fuel. This will allow for longer refuelling cycles and better plant capacity factors (reduced refuelling outages).

Figure 4.5 **Plutonium composition of 70%  $\text{ThO}_2$ -30%  $\text{UO}_2$  fuel burned to 72 MWd/kg**

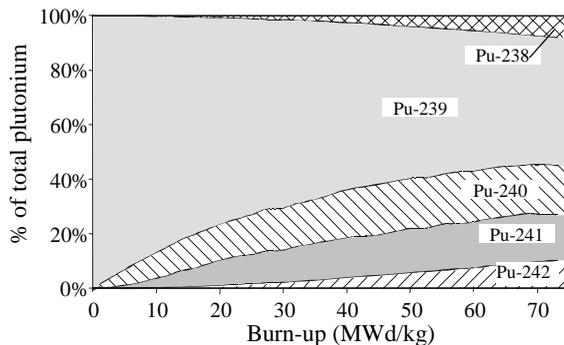
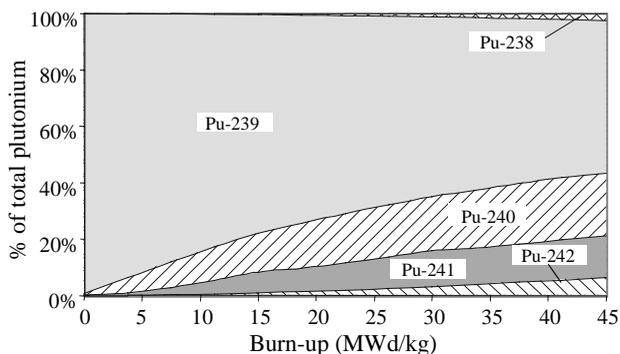


Figure 4.6 **Plutonium composition in a  $\text{UO}_2$  core burned to 45 MWd/kg**



Because  $\text{ThO}_2$  is the highest oxide of thorium, while  $\text{UO}_2$  can be oxidised further to  $\text{U}_3\text{O}_8$  or  $\text{UO}_3$ , spent  $\text{ThO}_2\text{-UO}_2$  fuel appears to be a superior waste form in the event that the spent fuel would ever be exposed to air or oxygenated water.

The main challenge encountered in the design of a thorium-based system is the necessity to supplement natural thorium with a fissile component, e.g.  $^{235}\text{U}$ . Several design solutions were proposed and investigated, such as the initial start-up of the thorium cycle by enriched uranium, the continuous addition of uranium as a fissile component to supplement self-generated  $^{233}\text{U}$ , the reprocessing and recycling of  $^{233}\text{U}$ , and the addition of plutonium to supplement self-generated  $^{233}\text{U}$ . Thorium, in metallic, nitride or oxide form, is being investigated as a host matrix for plutonium and minor actinides in an actinide burning reactor. As noted above, thoria has good thermal conductivity and fission gas retention properties. In addition, the use of thorium rather than uranium increases the net destruction of plutonium and higher minor actinides, because no additional  $^{239}\text{Pu}$  is being bred [95].

A significant improvement in natural uranium utilisation by using thorium could be achieved only if the self-generated  $^{233}\text{U}$  fissile material would be separated and recycled in a closed fuel cycle. However, the use of thorium and uranium in a once-through fuel cycle can result in increased burn-up and a more durable fuel/waste form. To reliably contain the fission products and actinides within the fuel after loss of the waste containers and cladding, the fuel must not undergo significant chemical reactions and resultant changes in crystal structure. Thorium dioxide is the highest oxide of thorium and does not depart significantly from its stoichiometric dioxide composition when exposed to air or water at temperatures up to 2 000 K [94]. Spent uranium dioxide fuel fragments, on the other hand, react at a rate of about 1% per year (i.e. relatively rapidly) with 90°C high-drip-rate water with representative Yucca Mountain contaminants [96]. Air-oxidation of  $\text{UO}_2$  spent fuel at temperatures near but below 200°C produces  $\text{U}_4\text{O}_9$  after only several years of exposure [97].  $\text{ThO}_2$ - $\text{UO}_2$  mixtures appear to be susceptible to corrosive attack in air or oxygenated water, but significantly less rapidly than  $\text{UO}_2$ . The particular molecular structure is important because the theoretical densities of  $\text{UO}_2$ ,  $\text{U}_4\text{O}_9$  and  $\text{U}_3\text{O}_8$  are 10.97, 11.4 and 8.35 t/m<sup>3</sup>, respectively. Therefore, conversion of  $\text{UO}_2$  to  $\text{U}_3\text{O}_8$  results in about a 30% volume increase, grain boundary separation, and powdering of the fuel. If the fuel is contained in typical LWR cladding, oxidation of the  $\text{UO}_2$  to  $\text{U}_3\text{O}_8$  is likely to cause severe splitting of the cladding [98]. However, oxidation to  $\text{U}_4\text{O}_9$  results in a slight densification (3.4%), some micro cracking, and no structural damage to the fuel. Cohen and Berman [99] investigated the extent of oxygen solubility in unirradiated  $\text{UO}_2$ - $\text{ThO}_2$  solid solutions as a function of temperature and composition. They found that after oxidation, the value of x in  $(\text{Th}, \text{U})\text{O}_{2+x}$  increased continuously from 0, at  $\text{ThO}_2$ , to a maximum value of 0.25 for 50% mixtures. In other words, heavily oxidised high thoria solid solutions, from  $(\text{U}_{0.5}\text{Th}_{0.5})\text{O}_{2+x}$  up to pure  $\text{ThO}_2$ , are in equilibrium with oxygen with urania molecular structures between  $\text{UO}_2$  and  $\text{U}_4\text{O}_9$  and, therefore, swelling, grain boundary separation, and fuel powdering does not occur. Higher urania oxides do not form when there is at least 50% thoria present because the thoria stabilises the fluorite structure and only one interstitial oxygen can be accommodated per unit cell, only where an appropriate number of uranium ions are adjacent to the space occupied by the oxygen. Thomas *et. al* [97] reported a somewhat similar stabilisation due to the fission products in high burn-up spent LWR  $\text{UO}_2$  fuel oxidised at temperatures below 200°C in air. The oxidation of  $\text{UO}_2$  in water requires the presence of dissolved oxygen, but “proceeds in a completely different manner from oxidation in air or gaseous oxygen” [100]. The major reaction product upon contact with pure water is  $\text{UO}_3 \cdot 0.8\text{H}_2\text{O}$ , a hydrate called dehydrated schoepite. Markowitz and Clayton [101] investigated the corrosion behaviour of a group of nuclear fuel oxides, including two compositions (20 and 50%  $\text{UO}_2$ ) of urania-thoria, in high temperature (360°C), alkaline (pH 10) flowing water. The  $\text{ThO}_2$  samples displayed excellent corrosion resistance. The weight gains of the urania-thoria material exposed to water oxygenated to about 100 ppm “were much larger than for any of the other materials tested, in any of the media, even those which failed and fell apart; yet all the specimens retained their mechanical integrity”. The urania-thoria fuel remained mechanically intact in spite of the growth of an oxidised surface phase thought to be of the  $\text{M}_4\text{O}_9$  type.  $\text{UO}_3 \cdot 0.8\text{H}_2\text{O}$  apparently was not found in or on the Markowitz and Clayton specimens.

Under irradiation, thorium undergoes a rapid increase in  $^{233}\text{U}$  concentration (and  $k_{\text{eff}}$ ) and retains a value higher than that of plutonium in uranium lattice. Because of the very large fission cross-section

of plutonium, the level of plutonium in uranium builds up (and  $k_{\text{eff}}$  increases) early in life, but these quantities saturate and decline much sooner than in case of  $^{233}\text{U}$  in thorium. This difference results from the fact that the equilibrium concentration of plutonium is low, by virtue of its high cross-section and because of the high capture to fission ratio.

For the U-Pu fuel cycle, the technology for the recovery and separation of uranium and plutonium from irradiated fuels is well established. The process (PUREX) is based on one particular version of the solvent extraction technique. A process suitable for thorium-based fuels is likely to have a broad similarity to the PUREX process, but with important differences. For example, the solvent extraction process in PUREX depends on the fact that the tetravalent plutonium is more highly soluble in tributyl phosphate (TBP) than is hexavalent uranium. For the separation of uranium and thorium, the process on which most work has been done is the Thorex process, which depends on the way that the relative solubilities of the two materials in nitric acid and in TBP vary with the acid concentration. Practical development of thorium reprocessing has not yet reached the stage where process details can be treated as definitive.

The processes for refabricating fuel containing recycled material would be expected to follow the same methods as for new fuel elements, but with an important difference. Recovered thorium contains  $^{228}\text{Th}$ , which decays with a half-life of 1.9 years through a series of hard gamma-emitting daughter products. This would cause severe handling problems during the refabrication of thorium fuel elements, making the reuse of recovered thorium unattractive after a long extra decay time. The recovered uranium isotopes, however, after removal of fission products will at first be essentially free from gamma activity but, because of the decay of the associated  $^{232}\text{U}$ , remote handling is essential unless refabrication can be undertaken within a few days after reprocessing. It is anticipated that refabrication plants will need extensive developments to adapt for remote handling, as they will be heavily shielded and maintenance will be complicated. It is to be noted that the requirement for shielding continues throughout the handling and transport route from the fabrication plant up to loading in the reactor.

The recycled  $^{233}\text{U}$  is also alpha active, creating a hazard if inhaled. Because of this, consideration is given, as with plutonium-fuelled fast reactors, to processes that minimise the generation of dust. There is, therefore, an interest in the development of gel precipitation processes to produce oxide spheres in a suitable range of sizes. These could form the basic fuel kernel for HTR coated particles, or could be loaded into metal-clad fuel pins for vibro-compaction.

### *Other R&D Programmes*

In addition to the activities discussed above, other important initiatives related to advancing the nuclear energy state-of-the-art are the Nuclear Energy Research Initiative in the USA the USA-led Generation IV International Forum (GIF) [102] and the IAEA-led International Project on Innovative Reactors and Fuel Cycles (INPRO). The goal of these initiatives is the design of one or more nuclear energy systems that can be licensed, constructed and operated in a manner that will provide a competitively priced supply of electricity while satisfactorily addressing nuclear safety, waste, proliferation and public perception concerns of the countries in which they are deployed. Additionally, several countries have programmes underway for evaluating the most promising candidates for next-generation reactors. These candidates include designs that make evolutionary improvements to existing concepts, as well as designs that are more innovative.

## 4.5 Conclusions

Three main conclusions can be drawn from the ongoing activities discussed above:

- Active research is underway on a number of new nuclear reactor and fuel cycle technologies.
- Most research is geared toward evolutionary improvement in reducing costs, increasing safety margins, improving post-disposal behaviour, and improving opportunities to manage increasing plutonium inventories.
- Research is being conducted on promising advanced concepts, but analysis and licensing of advanced fuels and technologies is a lengthy process that can take up to 15-20 years, meaning that innovative concepts are not likely to be introduced before the 2020 time frame.

## 5. FUEL CYCLE SCHEMES

### 5.1 Introduction

The previous chapters introduced today's developments in the fuel cycle and gave a brief overview of the main trends under consideration in the short to long term. As was stated in the introductory chapter, these developments should not be seen as stand-alone but rather as serving specific objectives aiming towards improving the economic, environmental and social dimensions of nuclear energy in the future. Ultimately, meeting these objectives could help to ensure that nuclear energy can play an important role as a sustainable energy source in the long term.

This chapter will describe a set of overall fuel cycle schemes that are considered relevant to this study. These fuel cycle schemes were selected based on the ongoing efforts in several OECD Member countries and on the proposed long-term focus of sustainability, i.e. delivering energy by effective use of resources at a competitive cost without harming the environment. In this respect, the chapter attempts to clarify for the reader the different options under consideration, and their merits and their drawbacks. Special emphasis is given to the technological maturity of the needed developments and the perceived impacts of technical developments in achieving a more sustainable nuclear option.

The set of fuel cycles includes 13 schemes<sup>60</sup> (see Table 5.1). These fuel cycle schemes have been grouped according to their maturity or perceived industrial potential in the short, medium and long term, in line with the technological maturity criteria introduced in Chapter 3 (see Tables 3.2 and 3.3). As a fuel cycle scheme is a combination of technological developments in various fuel cycle steps, each having its own status of technological maturity, its own pace of R&D progress and its own (economic) constraints, the ranking of the fuel cycle schemes according to the same four scales of technological maturity as for individual technologies would be misleading. Therefore, the broader rankings used in Table 5.1 are considered to be more appropriate for this study.

The following sections will detail these fuel cycle schemes and discuss their main merits and drawbacks as well as the underlying developments in deploying these schemes.

### 5.2 Mature fuel cycle schemes

Today's mature fuel cycles are essentially three. Their various steps were described in Chapter 2. These fuel cycle schemes will remain normatives for the coming ten to twenty years. However, increasing competition in the electricity sector makes it imperative for the nuclear business to constantly apply cost-reduction strategies, for both power plant investment costs and the fuel cycle operations. In fact, it is through the low nuclear fuel cycle costs that the higher investment costs of nuclear power plants, relative to fossil fuel-fired plants, can be compensated.

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60. The mass balances for the different fuel cycle schemes are analysed using the COSI-code (CEA), see Annex B.

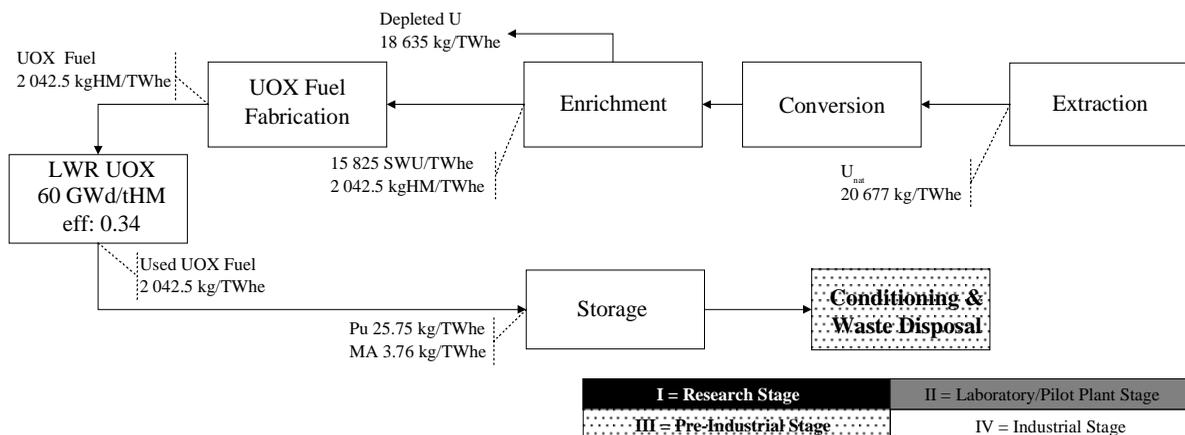
Figures 5.1 to 5.3 show the flow-sheets for these mature fuel cycle schemes,<sup>61</sup> together with the associated mass-flows. In addition, the technological maturity of each step of the fuel cycle is indicated through the shadings used for the boxes.

Table 5.1 Fuel cycles under consideration in this study

|   |
|---|
| <p><b>Currently Deployed On Industrial Scale</b></p> <p>OFC: Once-through fuel cycle<br/> HWR (CANDU): Once-through fuel cycle<br/> RFC: Recycle fuel cycle-monorecycling</p>   |
| <p><b>Potential Industrial Deployment in &lt;25 years</b></p> <p>DUPIC: Direct Use of PWR spent fuel In Candu<br/> HTGR: High-temperature gas-cooled reactors<br/> RFC-multirecycling<br/> LWR-Inert Matrix Fuel (IMF)<br/> LWR-Pu+MA recycling</p> |
| <p><b>Potential Industrial Deployment in 25-50 years</b></p> <p>Mixed LWR + FR fuel cycle<br/> FR 100% cycle<br/> Double strata P&amp;T fuel cycle<br/> Thorium-cycle<br/> Molten salt fuel cycle</p>   |

The once-through fuel cycle (Figure 5.1) serves as the reference fuel cycle in operation in most of the OECD Member countries. Today’s low uranium prices, an over-capacity in enrichment and fuel fabrication and higher levels of irradiation possible per fuel element, enable utilities to continuously optimise the economic dimension of this fuel cycle.

Figure 5.1 Once-through fuel cycle (OFC)



61. The details and parameters used in the flow-sheet calculations are specified in Annex B.

Figure 5.2 CANDU fuel cycle (PHWR)

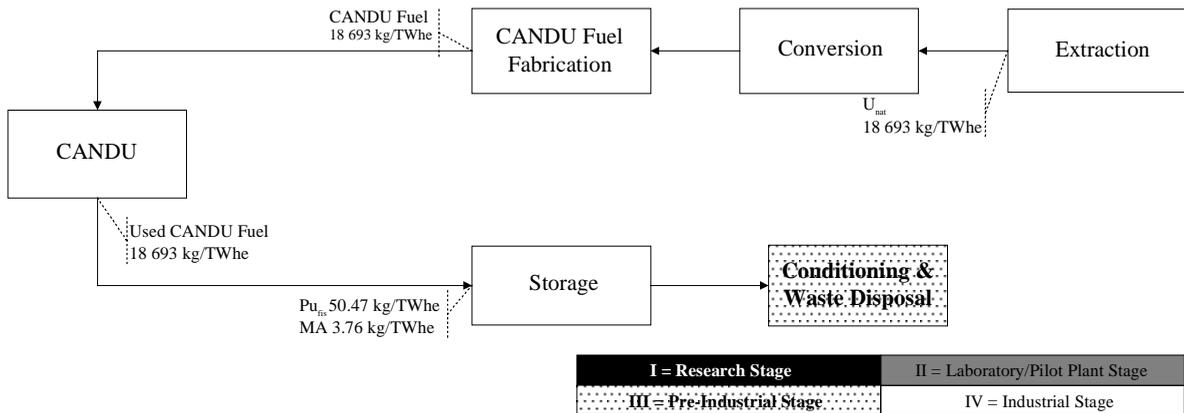
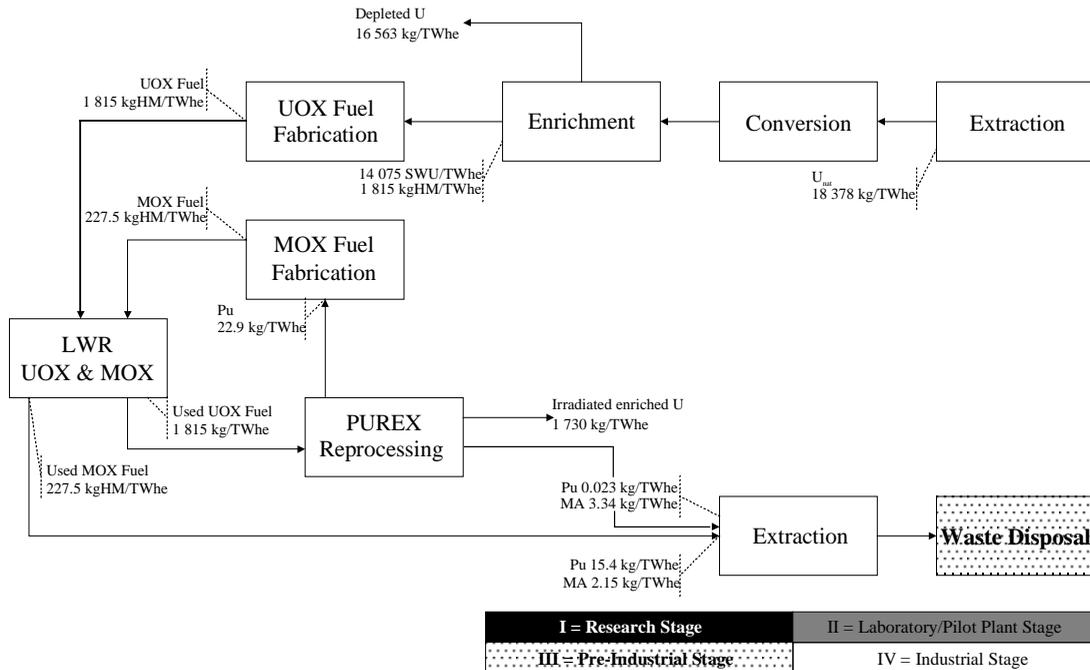


Figure 5.3 Plutonium mono-recycling fuel cycle (RFC)



As shown in the flow-sheets, all stages in the three fuel cycle schemes are industrially mature, except for the final stage in the fuel cycle, i.e. waste disposal. The final disposal of spent fuel or high-level wastes from reprocessing has not yet been performed on an industrial scale in OECD Member countries and only interim storage of spent fuel (for periods up to several decades) has been industrially implemented, as wet or dry storage either at the reactor site or at remote sites. The major constraints for effective disposal relate to socio-political issues rather than to technical problems.

The past ten years have shown steady technological progress in the realisation of geological disposal. However, the technological progress was oriented mainly towards continuous improvement of the disposal techniques and a better understanding of the scientific and technical uncertainties. Progress towards realisation of operational geological disposal sites is delayed by socio-political constraints, e.g. stakeholder confidence. In certain countries (e.g. USA, Germany), this delay translates also into a delay to transport the spent fuel from the reactor sites or to return the high level waste from

(foreign) reprocessing sites to the national geologic disposal site. This results in additional costs to the utilities for the storage of spent fuel or reprocessing wastes.

As will be introduced later in this chapter, additional waste management options, such as partitioning and transmutation, have been proposed but their development should be seen also within the socio-political context. Despite today's lack of general public acceptance to realise geological disposal sites, the development and realisation of such sites only needs a few decades. Partitioning and transmutation (P&T) could partly reduce the mass and radiotoxicity of some long-lived isotopes going to waste by a factor of at least hundred. However, the deployment of P&T systems and achievement of these reduction factors will require decisions for their development and also commitment to use such advanced fuel cycle options over a period of at least hundred years.

The once-through fuel cycle for HWRs (see Figure 5.2) is somewhat simpler than that for LWRs, because no enrichment step is involved. The main difference, besides the elimination of the enrichment step, is the low burn-up (typically between 7.5 and 15 GWd/tHM) of the HWR fuel and, as a consequence, the higher amount of spent fuel per TWhe of electricity production. The low burn-up of the fuel also makes the fissile content of plutonium formed during irradiation rather high. The decay heat of the spent fuel is lower than in the high-burn-up LWR case and does not create any specific problems. In order to increase the burn-up of the HWR fuel, and consequently increase the energy produced per unit of fuel, the use of slightly enriched uranium (SEU) or even reprocessed uranium,<sup>62</sup> is being considered. Uranium enrichment up to about 1.2% would enable burn-ups higher than 15 GWd/tHM.

Compared to the once-through fuel cycle for LWRs, the plutonium recycling fuel cycle (see Figure 5.3) achieves a saving of about 12% in natural uranium requirements if only the plutonium recovered during reprocessing is recycled. This savings increases to about 20% if the recovered uranium would be recycled also.<sup>63</sup> In the recycle fuel cycle (RFC), the irradiated spent UOX fuel is first stored for about four years to allow the decay of residual heat and radioactivity of the short-lived fission products, and the spent fuel is then reprocessed to recover the plutonium and uranium. Today, only one recycle of recovered plutonium is implemented in LWRs. However, a second recycling is technically feasible with few changes in existing reactors, particularly with regard to reactivity control systems. Beyond two recyclings, some safety parameters, such as void coefficient, may be significantly affected because of changes in the plutonium isotopic composition which requires higher Pu concentrations in MOX fuel assemblies. As a result, specific and expensive modifications to fuel and core design or reactivity control systems would be needed.

While the spent UOX fuel typically has a residual decay heat of about 2 kW/tHM after 4 years cooling, spent MOX fuel with comparable burn-up has a residual decay heat of about 8 kW/tHM (3 to 4 times more and slowly decaying) which imposes longer storage times before further conditioning of the spent MOX fuel could be started. The industrial feasibility of reprocessing spent MOX fuel has been demonstrated with campaigns carried out in France at Marcoule and La Hague in 1992 and 1998, during which more than 10 t of MOX fuel were successfully reprocessed. However, the reprocessing of MOX fuels on a large scale would require some modifications of existing UOX fuel reprocessing facilities, e.g. to the head-end operations.

It is thought that developments in reprocessing technology could enable a reduction of the reprocessing cost by at least 30%. In the RFC, currently reprocessing accounts for about 30% of the fuel cycle cost; thus, this cost reduction would reduce the fuel cycle cost by some 9%. The cost reducing developments include further streamlining and improvement of the processes, reduction of the number of

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62. This would however introduce <sup>236</sup>U into the fuel.

63. While such savings can be important, one should not forget that less than 1% of the energy content of the mined uranium is used for energy production in current reactors and fuel cycles.

process steps and waste streams, converting batch processes to continuous processes where possible, and reduction of the overall size of the equipment and plant. Fuel shearing and associated maintenance facilities have a large minimum size which is a determining factor on economics of scale and viable industrial capacity. Batch dissolution of fuel requires large buffer capacity and creates secondary waste streams. Two major lines of research being pursued focus on intensification of the head-end processes: development of dissolvers in which the fuel and hulls are transported, during the dissolution process, by liquor pulses and by centrifugal contactors; and an electrochemical process which combines the now separate cladding removal (shearing) and dissolution steps.

A possible discovery of  $\text{PuO}_{2+x}$  compounds, having higher mobility due to higher solubility in water, could have an impact on the disposal of Pu-containing spent fuel in a geological disposal [103]. In addition, this would increase the importance of reduced Pu production and could lead to the requirement that the fuel matrix also act as a robust waste-form in the OFC. However, other experiments made, e.g. France, could not yet reproduce higher oxidation states for Pu than  $\text{PuO}_2$ , leading to the conclusion that this might not happen in realistic disposal conditions.

Although the choice between these three fuel cycles rely mainly on socio-political criteria, which remain country-specific, one may conclude that all three cycles respond successfully to the need for economic, environmentally friendly and safe energy production and that developments, as will be discussed in the remainder of this chapter, focus on improving these fuel cycles to better meet each nation's economic, socio-political and technological needs.

Chapter 3 introduced the criteria and indicators that are considered relevant in the analysis of the sustainable character of current and future nuclear energy production. While certain of these criteria and indicators remain invariant to the developments described here, some of the criteria do condition the technological feasibility of improving the whole set of criteria. Figure 5.4 summarises the performance of these three mature fuel cycles according to some prime criteria, i.e. use of natural resources, waste arising, radioactivity of waste, collective dose to workers and public, etc.

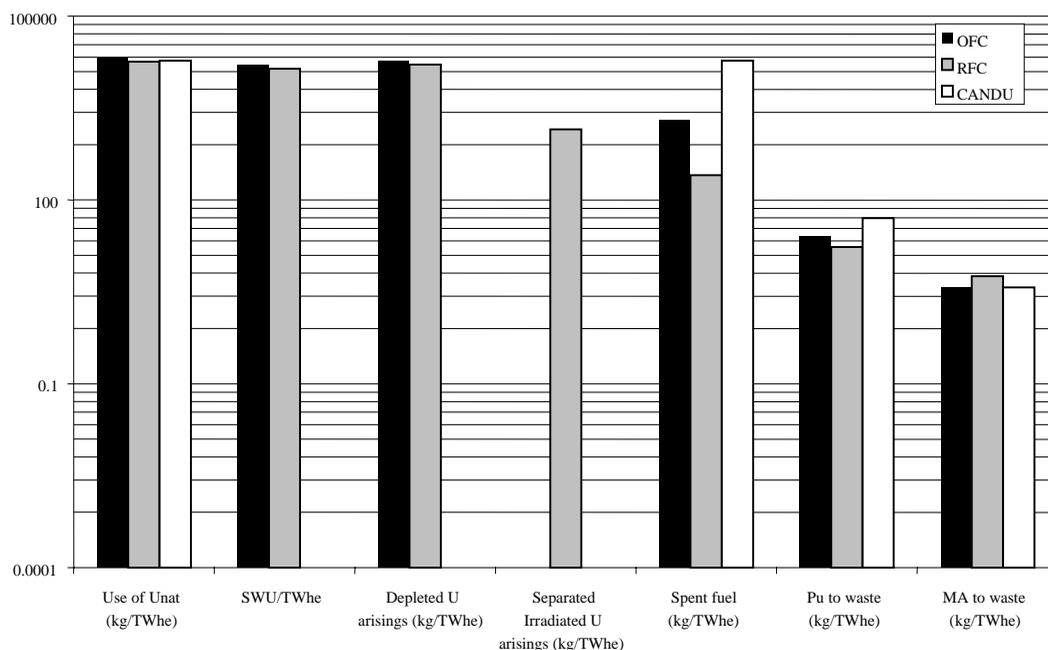
While this graph could lead to comparison between these fuel cycles, one must keep in mind that the whole fuel cycle should be seen as an integrated system, in which a higher value for one indicator can well be compensated by lower values for other indicators. The remainder of this chapter will analyse how developments may impact the performance of each of these fuel cycles while showing that trade-offs are needed in order to improve the fuel cycle as a whole.

### **5.3 Potential industrial fuel cycles schemes in the short to medium term**

The short-term developments will be characterised by the goal to reduce fuel cycle costs, based essentially on some evolutionary changes to today's practices, and no decisions to invest in radically new technologies are envisaged.

In the medium time-frame, one may expect developments aiming, in particular, towards solutions improving the back-end of the fuel cycle, i.e. management of transuranics in the spent fuel. Other possible developments will aim at adapting the fuel cycle to some specific renewed interest in former reactor technologies, e.g. renewed interest in high-temperature gas-cooled reactors, that allow nuclear energy to be used beyond electricity production only.

Figure 5.4 Performance of mature fuel cycles (OFC, RFC and CANDU) according to a selected set of indicators



### 5.3.1 Plutonium multi-recycling scheme and MIX scheme

Chapter 4 already introduced some of the fuel concepts that could be developed for the management of plutonium in more advanced fuel cycles. Where today's RFC, as deployed in France, could result in a stabilisation of the separated Pu stock, a decrease of this stock would be feasible only by increasing the Pu content in the reactor core either by increasing the Pu content in the MOX fuel and/or increasing the MOX fraction up to 100% in LWRs. As explained in the previous chapter, only high-moderation reactors (HMR) and MIX concepts seem to be feasible options while using existing or slightly modified LWR technology.

Recycling plutonium in standard LWRs using 100% MOX reload would bring about a degradation of the control system effectiveness. In order to avoid these effects, modifications to the reactor or to the fuel cycle must be implemented. The HMR reactor option relies on changes in the reactor by introducing high-moderation fuel assemblies. If one would envisage, however, a multi-recycling of the plutonium in existing or only slightly modified LWRs, then the MIX option seems to be more appropriate. While the HMR option is limited to about two recyclings in standard LWRs, and therefore still needs Pu recovered from UOX fuel, this limitation does not exist for MIX fuel as the degradation of the Pu isotopic mix with multiple recycling is reduced. The flow scheme for such a fuel cycle is shown in Figure 5.5. A variant of this scheme includes also recycling of minor actinides (MA) in the MIX fuel as shown in Figure 5.6.

The main difference between the MOX fuel cycle option (RFC) and the MIX option is the higher Pu inventory in the fuel. In the MOX option with only once or twice recycling of Pu, the spent fuel contains about 15.4 kgPu/TWhe which leaves the fuel cycle for geological disposal, whereas Pu is recycled in the MIX option. The multiple recycling of Pu in the MIX fuel supposes, however, a major increase in MIX fuel fabrication capacity (2 042 kg/TWhe compared to 227 kg/TWhe in the MOX option) while the demand on reprocessing capacity remains the same. The MIX option relies on

today's technologically mature processes, and no specific technical problems seem to be encountered for this extension. One should note the rather large reduction in Pu going to waste disposal while, in contrast, the amount of MA going to waste increases by a factor of about 2 (8.7 kg/TWhe instead of 5.5 kg/TWhe). In order to cope with the accumulation of minor actinides, it can be envisaged to recycle them in the MIX-MA cycle (see Figure 5.6). Such a MA-recycling in LWRs would concentrate the MAs in the fuel cycle, but the MA discharges to waste disposal eventually would be reduced by a factor of over 100 (0.018 kg/TWhe versus 8.7 kg/TWhe).

Figure 5.5 MIX fuel cycle without MA recycling

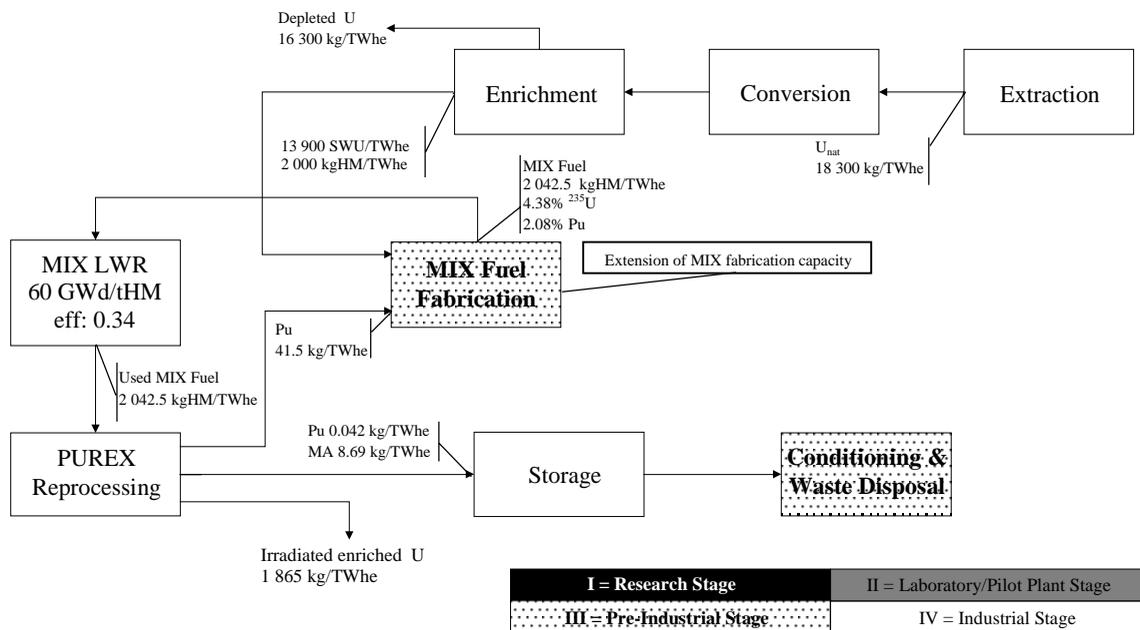


Figure 5.6 MIX fuel cycle with MA recycling

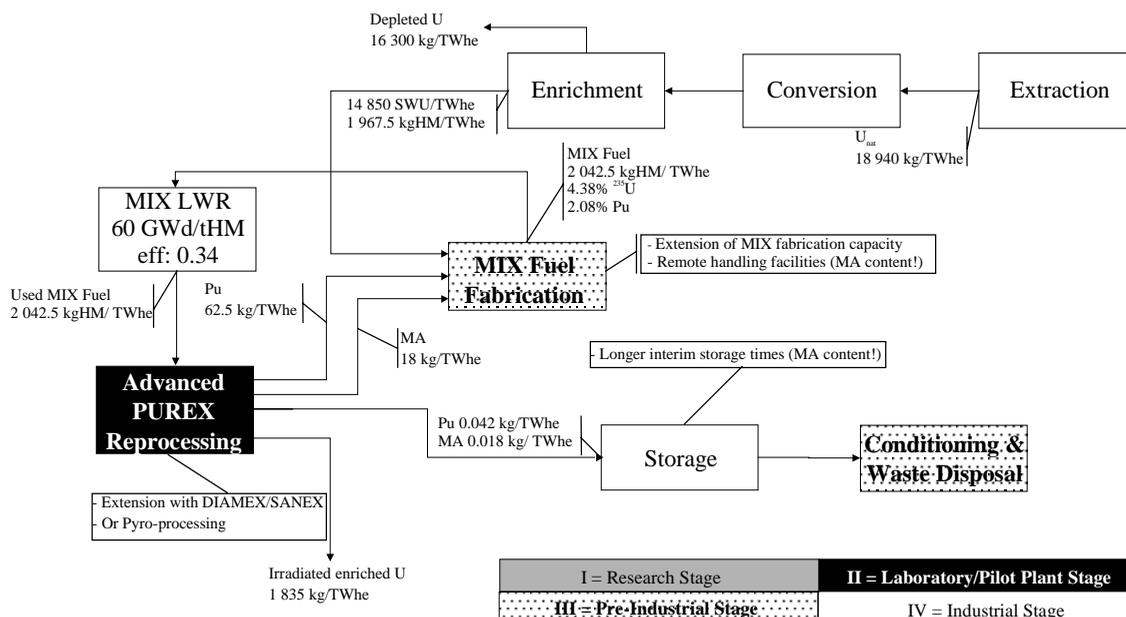
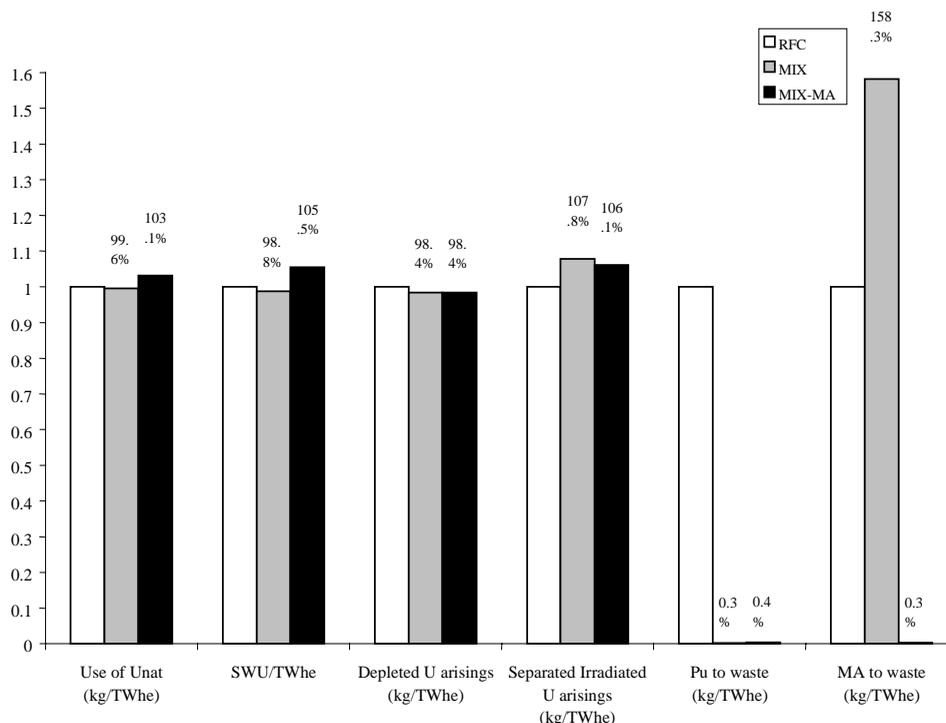


Figure 5.7 Comparison of selected criteria for fuel cycle schemes RFC, MIX and MIX-MA



Some issues in fuel fabrication are encountered when minor actinides would be included in the mixed oxide fuel. The presence of  $^{232}\text{Np}$  does not affect the alpha and neutron emission but increases the gamma source, due to  $^{233}\text{Pa}$ . In case of multi-recycling of Np,  $^{238}\text{Pu}$  is progressively built-up, and this additional source of neutrons and heat (mostly from  $(\alpha, n)$  reaction) further affects the reprocessing and refabrication steps. The addition of americium to the U-Pu powder for recycling would induce a strong increase of the gamma dose (by a factor of 4.5). The main problem is, however, related to the curium to be included in the fuel. The addition of Cm to MOX fuel, for example, would increase the neutron dose rates around the blending glove boxes in the fabrication process by about a factor of 100. This would require thick protection layers (0.3 to 1 m polyethylene shields) similar to those in reprocessing facilities, leading to sophisticated remote fabrication lines and thus resulting in a rather expensive process [75].

Therefore, if recycling of minor actinides is considered to be of fundamental importance, the LWR seems not to be an optimal solution compared to incineration in fast reactors and limiting the MA-containing fuel to a smaller part of the fuel cycle. In any case, the use of targets instead of homogeneous MA-containing fuel is to be envisaged (see Section 4.2.3.2.2, inert matrix fuels).

Figure 5.7 shows the quantitative data for these fuel cycles in comparison to the RFC. Only minor changes appear in the front-end of the fuel cycles, the major differences being in the back-end and especially in the characteristics of the wastes going to geological disposal.

### 5.3.2 The DUPIC fuel cycle

An extension of the current OFC-LWR cycle could be the addition of an “after-burn” of the LWR spent fuel in a HWR (CANDU) type of reactor. The good neutron economy of this type of reactor

would enable the use of the spent UOX fuel<sup>64</sup> in a DUPIC<sup>65</sup> fuel cycle (see Figure 5.8) which would achieve an additional 15 GWd/tHM burn-up. The spent DUPIC fuel would be sent to interim storage and final disposal.

The heart of the DUPIC fuel cycle is the OREOX (Oxidation and Reduction of OXide fuels) process for transforming the LWR spent fuel into new fuel for the HWR. AECL and KAERI have conducted some experimental tests with the OREOX process, and have made test irradiation of the produced fuel. During the dry OREOX process, uranium from spent LWR fuel is sequentially oxidised and reduced to a fine powder, which forms the starting material for fabrication of DUPIC fuel pellets. The powder is pressed into pellets, sintered to a high density, ground to final size, and seal-welded within Zircaloy sheaths. Because the fuel remains highly radioactive, all processing steps should be carried out in shielded hot cells. During this process, volatile fission products escape from the spent fuel and are trapped in ceramic-based filters. The <sup>137</sup>Cs is the most radioactive and should be stored until its decay heat has decreased significantly (approximately 50 to 100 years) before going into disposal.<sup>66</sup> The other off-gases collected from the OREOX process generate no heat, but contain very long-lived radio-nuclides (e.g. <sup>129</sup>I, <sup>99</sup>Tc and <sup>14</sup>C) requiring deep geological disposal. The final DUPIC fuel assembly is rather simple in design and small (0.5 m long) compared to an LWR assembly, which could facilitate remote fabrication.

The use of the DUPIC fuel cycle option would result in about a 3-fold reduction in the quantity of spent fuel arisings per unit electricity generated, compared with direct disposal in a dual open operating system, in which CANDU reactors are fuelled with natural uranium and LWRs are fuelled with enriched uranium. Countries having both type of reactors potentially could use this option if the process would become industrially operational. Questions do arise about the impact of higher burn-up LWR fuel on the potential of the DUPIC option, and especially its economic viability. The current trend with UOX/MOX fuel is to higher burn-ups (up to about 70 GWd/tHM), resulting in higher contents of minor actinides, which could seriously hamper the OREOX process.

The DUPIC development programme includes Korea (KAERI), Canada (AECL) and the USA (DOE and LANL), and was started in 1991 by conducting some feasibility studies that ended with the selection of the OREOX process. The demonstration of the OREOX process has since then been started on a laboratory scale with subsequent irradiation in research reactors. A total of 3 DUPIC elements were fabricated in the hot cells at AECL's Whiteshell Laboratories Shielded Facilities. Approximately 3 kg of spent PWR fuel with an average burn-up of 28 GWd/tHM was processed into fuel pellets using the OREOX process, then the pellets were formed into stacks, loaded into fuel elements and welded. Irradiation testing of the DUPIC elements in the NRU reactor up to 21 GWd/tHM started in 1999 March. In Korea, KAERI plans experimental testing of DUPIC fuel in the HANARO reactor.

The development of the DUPIC fuel cycle, and especially the OREOX process, up to industrial level is estimated to require a further 10 to 15 years, depending on the economic interest for such a fuel cycle. With today's increasing LWR fuel burn-ups, the added-value of an "after-burn" in a CANDU is diminished and this makes the OREOX process even less attractive. Only those countries operating both LWR and CANDU type of reactors could take advantage of the potential synergy, but no economical and technological assessment of these conditions has been reported yet. The DUPIC fuel fabrication is reported to cost about 430-800 USD/kgHM, which is similar to current MOX fuel

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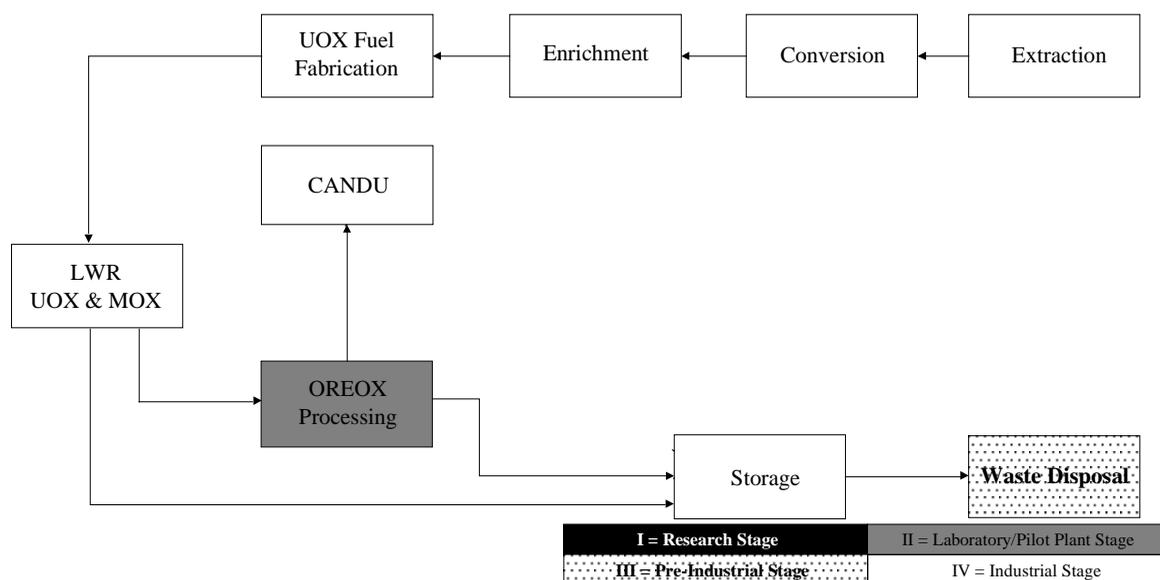
64. However, with rather low burn-up of only about 35 GWd/tHM.

65. Direct Use of PWR spent fuel in CANDU.

66. Preliminary unpublished analysis indicate that the co-disposal of caesium with the spent DUPIC fuel would increase the disposal costs by about 10%.

fabrication cost [90], and similar also to reprocessing and to spent fuel conditioning costs, respectively, but some ten times higher than for natural uranium CANDU fuel fabrication (47-83 USD/kgHM). A probabilistic simulation method was applied by Ko [90] to analyse the uncertainties in the DUPIC fuel cycle cost for the case of a possible Korean nuclear power park consisting of LWRs and CANDUs. The analysis showed that the savings in natural uranium resources, compared to a OFC for the LWRs, would amount to about 5% for a ratio of LWRs to CANDUs of 7, while these savings would become around 30% for a ratio of about 2. Spent fuel amounts would be reduced by 16% and 44%, respectively. The analysis finally indicated that the fuel cycle costs for the DUPIC cycle, for both cases of differing number of LWRs and CANDUs, would be equal to, or less than, the LWR once-through fuel cycle cost. Questions do arise if the assumed costs (430-800 USD/kgHM) for the DUPIC fuel fabrication are not achievable; however, the same analysis showed a low fuel cycle cost sensitivity to this fuel fabrication cost.

Figure 5.8 DUPIC fuel cycle



### 5.3.3 High-temperature gas-cooled reactor fuel cycles

The basic fuel cycle for a high-temperature gas-cooled reactor is an open fuel cycle scheme where, based on the very good fission product retention potential of the coated fuel particles, direct disposal of the fuel can be envisaged. However, it must be noted that former HTGR fuel cycle concepts did consider the Th-U cycle using highly enriched U, and reprocessing was foreseen in order to recover  $^{233}\text{U}$  and  $^{235}\text{U}$ . Chapter 4 has introduced most of the aspects related to fabrication of such HTGR fuel. The main difference relates to the very high burn-up (approaching 700 GWd/tHM) compared to the LWR-case (45-65 GWd/tHM). This, of course, results in a drastic drop in use of natural uranium and of the amount of spent fuel going to final disposal. Where a LWR-OFC uses 2 042 kilograms of natural uranium per TWhe, a HTGR (with 130 GWd/tHM burn-up) uses 668 kg/TWhe while a HTGR operating at 600 GWd/tHM would use only 145 kg/TWhe. The amount of plutonium produced drops from 26 kg/TWhe (LWR-OFC) to 15.3 for a HTGR (130 GWd/tHM) and the MA content drops from 3.76 kg/TWhe to 2.3 kg/TWhe.

In Japan, JAERI has considered several back-end options for the HTTR fuel, including the possibility of reprocessing. Reprocessing the HTTR-fuel was studied during the late 70s and early 80's, but is not being carried out at present. Some head-end reprocessing techniques were investigated

for applying the conventional PUREX process to the HTTR fuel. JAERI also studied graphite-CO<sub>2</sub> reaction and jet grind methods as head-end reprocessing techniques [104].

## 5.4 Fuel cycle schemes for the long term

While the short and medium term developments in the fuel cycle relate mainly to the economic dimension as well as trying to secure the technological knowledge for new nuclear applications (e.g. Pu burning, MA burning), some more advanced fuel cycle concepts also have been studied theoretically or on a laboratory scale during the past years, that could not be deployed within the next twenty years. Whereas most of the developments in the front-end, described in Chapter 4, apply to all fuel cycle schemes, most of the new concepts explicitly focus on the back-end and aim especially at dealing with the remaining waste. In principle, the approach taken was three-fold:

- Introducing additional waste management options, such as partitioning and transmutation, in order to reduce the mass and radioactivity of wastes going to final disposal, i.e. closing the fuel cycle not only for plutonium but also for the minor actinides. These schemes (LWR/FR and double strata) still rely on LWR technology in the first stratum and the fast reactor systems are designed as TRU or MA burner systems.
- Secondly, going to fully integrated cycles, such as 100% fast reactor cycles or even molten salt fuel cycles, which combine higher reactor thermal efficiency with optimising the use of natural resources and reducing the waste going to final disposal.
- Finally, changing the fuel from the U/Pu cycle to a thorium cycle where the long-term impact of long-lived radio-nuclides is altered. Different concepts have been proposed in this line, using modified LWRs, FRs, HTRs or MSR.

The following sections will highlight some of these concepts while indicating where the main advantages would be identified and showing the long-term dimension of these concepts. One should keep in mind that most of the available data are based on estimates which are mostly extrapolations of today's laboratory scale experiments or calculations and, therefore, precise values and sometimes even key assumptions are difficult to determine.

### 5.4.1 Fast reactor fuel cycle schemes

The previous sections have already indicated that a more sustainable use of uranium resources, while minimising the amount of long-lived radioactive waste produced, can be achieved by the use of fast spectrum systems. As most of the transuranics become fissionable in fast neutron spectra, the transuranics also contribute to the energy production while less of them will go to waste.

Initially, the FRs were designed as net breeders, i.e. producing more fissile material than is burned in the reactor. However, as demand for nuclear energy slowed and as more low-cost uranium became available, the main driving force for breeder reactors weakened. The R&D on FRs therefore evolved from their use as breeders to use as burners of plutonium. Several countries, e.g. Japan, Russia and China, still have an active FR programme while others, e.g. France and the USA, have slowed or stopped, at least temporarily, active development. However, new aggressive R&D programs are examined in the GEN-IV program and other international discussions.

FRs can be designed as actinide burners in a fuel cycle scheme combining LWRs and FRs. In this case, the actinides produced by irradiating the fuel in LWRs are burnt in FRs. As a TRU burner, a FR can burn only Pu (e.g. CAPRA-type of reactor) or both Pu and MAs (e.g. IFR-type of reactor). Burning MAs can be done either homogeneously in the fuel, or heterogeneously by fabricating dedicated targets, which can be then reprocessed or directly disposed. These options are illustrated in Figures 5.9 and 5.10.



The economics of advanced FR parks is a key factor in realising the commercial utilisation of the FR recycle system in the future. Some studies have been made by JNC [105] on the economic aspect of such fuel cycles. The target for the electricity generation cost of a future FR system commencing operation in the year 2030 was taken to be equal or lower than that of the future LWR. Table 5.2 summarises the cost estimates for current and future FR fuel cycle systems where use of advanced technologies is envisaged, e.g., wet MOX reprocessing technology (having low decontamination factors) and a fuel fabrication process with gelation and vibration packing technology considered to be both installed in one building in order to integrate the fuel cycle steps on one site.

**Table 5.2 Estimated costs for current and future FBR designs and associated advanced fuel cycle [106]**

| <b>Item</b>                              | <b>Current</b> | <b>Future</b> |
|--|----------------|---------------|
| Fuel burn-up (GWd/tHM)                   | 80             | 150           |
| Construction cost of a reactor (USD/kWe) | 4 700          | 2 000         |
| Reprocessing cost (USD/kgHM)             | 7 800          | 2 700         |
| Fuel fabrication cost (USD/kgHM)         | 4 900          | 1 600         |

Table 5.3 shows various characteristic values for the different scenarios, which can be used for a quantitative comparison for each step of the cycle, based on the criteria defined in Chapter 3. One sees very clearly, for example, the reduction in the amount of actinides contained in the waste for the recycling scenarios. Figure 5.12 shows a graphical representation of these data.

The so-called “Double Strata” fuel cycle scheme is an extension of the mixed LWR+FR fuel cycle scheme where all the separated minor actinides are treated in a second stratum and are transmuted by use of an accelerator-driven system (ADS). Figure 5.13 shows this type of fuel cycle including a CAPRA-type of fast reactor to burn the plutonium coming out of the UOX/MOX fuel cycle steps, and with partitioning and transmutation of the minor actinides from the mixed LWR+FR fuel cycle takes place in the second part of the cycle.

The advantage of such a double strata scheme relates to the concentration of the hazardous highly radioactive radio-nuclides in a separate part of the fuel cycle where dedicated fast reactor systems, such as ADS, can transmute them. In addition, while such dedicated transmuters and associated fuel cycle are very expensive operations, they constitute only a small part of the nuclear park (about 5 to 8%), and calculations have indicated that the levelised fuel cycle cost would increase only by about 15% [107]. The performance of such a fuel cycle in reducing the mass and radioactivity is comparable to an Integral Fast Reactor (IFR) park. The ADS in such a Double Strata scheme would, however, enable to reach the equilibrium sooner as well as to burn the remaining waste faster than using FRs alone in an IFR-scheme.



Table 5.3 Fuel cycle characteristics

|                                     | Once-through<br>OFC-LWR | CANDU<br>OFC-HWR | Monorec.<br>RFC-LWR | MIX<br>multirec.<br>RFC-MIX | MIX with<br>MA multirec.<br>RFC<br>Pu&MA-MIX | Mixed<br>LWR+FR<br>FC | Mixed<br>LWR+FR<br>FC with<br>AmCm targets | Pure FR park | Double-strata<br>FC |
|-------------------------------------|-------------------------|------------------|---------------------|-----------------------------|--|-----------------------|--|--------------|---------------------|
| Burn up (GWd/tHM)                   | 60                      | <15              | 60                  | 60                          | 60   | 60 & 150(FR)          | 60 & 145(FR)                               | 123          | ?                   |
| efficiency                          | 0.34                    | ?                | 0.34                | 0.34                        | 0.34   | 0.34 & 0.42           | 0.34 & 0.40                                | 0.40         |                     |
| Park (%) UOX                        | 100                     | 100              | 89                  | 0                           | 0  | 56                    | 44   | 0            | 65                  |
| MOX-LWR                             | 0                       | 0                | 11                  | 0                           | 0  | 0                     | 0  | 0            | 9.5                 |
| MIX-LWR (or ADS)                    | 0                       | 0                | 0                   | 100                         | 100  | 0                     | 0  | 0            | 6 (ADS)             |
| MOX-FR                              | 0                       | 0                | 0                   | 0                           | 0  | 44                    | 56   | 100          | 19.5                |
| Enrichment UOX <sup>235</sup> U (%) | 4.9                     | <1.2             | 4.9                 | --                          | --   | 5.3                   | 4.9  | --           | ?                   |
| Enrichment (%)                      |                         |                  |                     |                             |  |                       |  |              |                     |
| MOX/MIX <sup>235</sup> U            | --                      | --               | 0.25                | 4.38                        | 4.52   | 0.3                   | 0.25                                       | 0.25         | ?                   |
| Pu tot                              | --                      | --               | 10                  | 2.08                        | 2.82   | 23.3(core)            | 22.7(core)                                 | 19.9 core    | ?                   |
| MA                                  | --                      | --               | 0                   | 0                           | 0.86   | 0                     | 0  | 0            |                     |
| Unat (t/TWh)                        | 20.7                    | 18.7             | 18.4                | 18.3                        | 18.9   | 13.8                  | 9.4  | 0.7(depl.)   | 11.95               |
| Enrich. (SWU)                       | 15825                   | 0                | 14075               | 13900                       | 14850  | 8808                  | 7000                                       | --           | ?                   |
| Fab.(t/TWh)                         | 2.04                    | 18.7             | 1.82                | 0                           | 0  | 1.13                  | 0.905                                      | 0            | 1.58                |
| UOX                                 | 0                       | 0                | 0.23                | 0                           | 0  | 0                     | 0  | 0            | 0.23                |
| MOX-LWR                             | 0                       | 0                | 0                   | 2.04                        | 2.04   | 0                     | 0  | 0            | 0.037(ADS)          |
| MIX (or ADS)                        | 0                       | 0                | 0                   | 0                           | 0  | 0.28                  | 0.393                                      | 0.84         | 0.11                |
| MOX-FR                              | 0                       | 0                | 0                   | 0                           | 0  | 0                     | 0.0039                                     | 0            | 0                   |
| Am-Cm targets                       |                         |                  |                     |                             |  |                       |  |              |                     |
| Reproc.(t/TWh)                      | 0                       | 0                | 1.82                | --                          | --   | 1.13                  | 0.905                                      | --           | 1.58                |
| UOX                                 | --                      | --               | 0                   | --                          | --   | --                    | --   | --           | 0.23                |
| MOX-LWR                             | --                      | --               | --                  | 2.04                        | 2.04   | --                    | --   | --           | 0.037(pyro)         |
| MIX (or ADS pyro)                   | --                      | --               | --                  | --                          | --   | 0.28                  | 0.393                                      | 0.84         | 0.11                |
| MOX-FR                              |                         |                  |                     |                             |  |                       |  |              |                     |
| Wastes (kg/TWh)                     | 25.8                    | 50.5             | 15.4                | 0.04                        | 0.06   | 0.06                  | 0.24                                       | 0.14         | 0.09                |
| Pu                                  | 3.76                    | ?                | 5.5                 | 8.7                         | 0.018  | 4                     | 0.28                                       | 4.3          | 0.01                |
| MA                                  | 1.87                    |                  | 1.71                | 1.68                        | 0.003  | 1.25                  | 0.003                                      | 0.43         |                     |
| Np                                  | 1.61                    |                  | 3.08                | 4.62                        | 0.007  | 2.12                  | 0.04                                       | 3.58         |                     |
| Am                                  | 0.28                    |                  | 0.72                | 2.37                        | 0.009  | 0.55                  | 0.235                                      | 0.28         |                     |
| Cm                                  | 125.4                   |                  | 125                 | 124.2                       | 120.3  | 108.6                 | ?  | ?            | 6.5                 |
| PF                                  |                         |                  |                     |                             |  |                       |  |              |                     |

Figure 5.12(a) Comparison of fuel cycle schemes according to the fuel cycle service demands

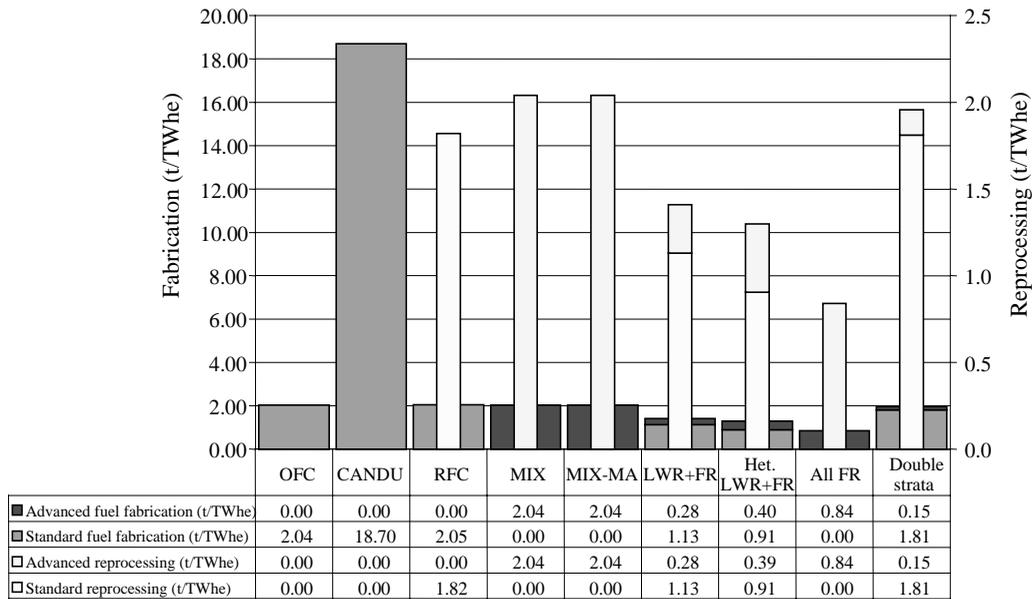


Figure 5.12(b) Comparison of fuel cycle schemes according to the amounts of TRU waste going to final disposal (kg/TWhe)

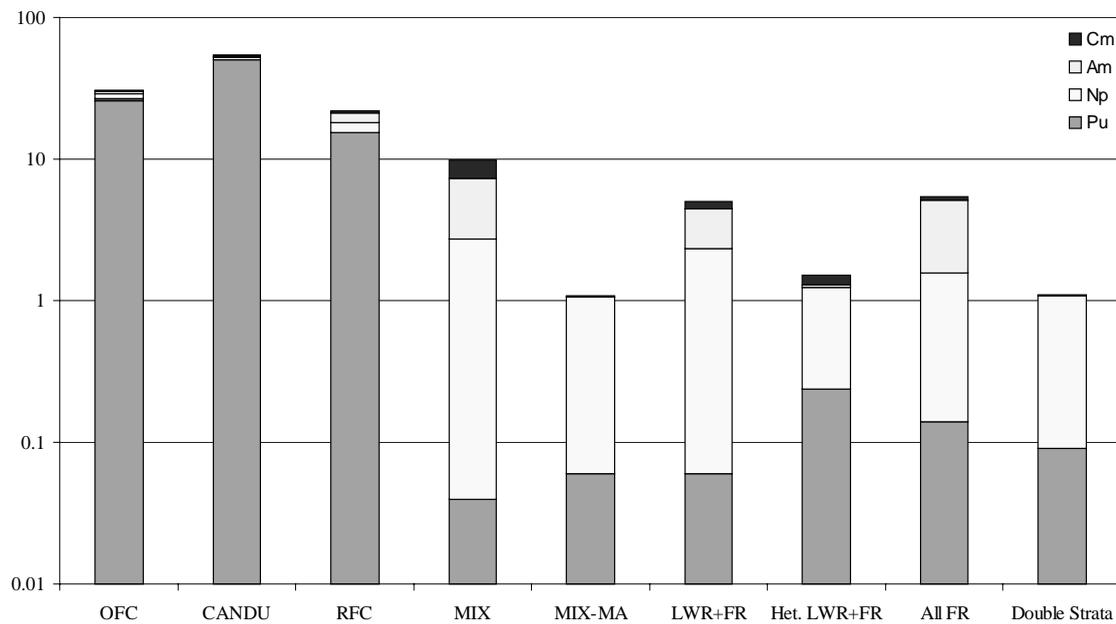
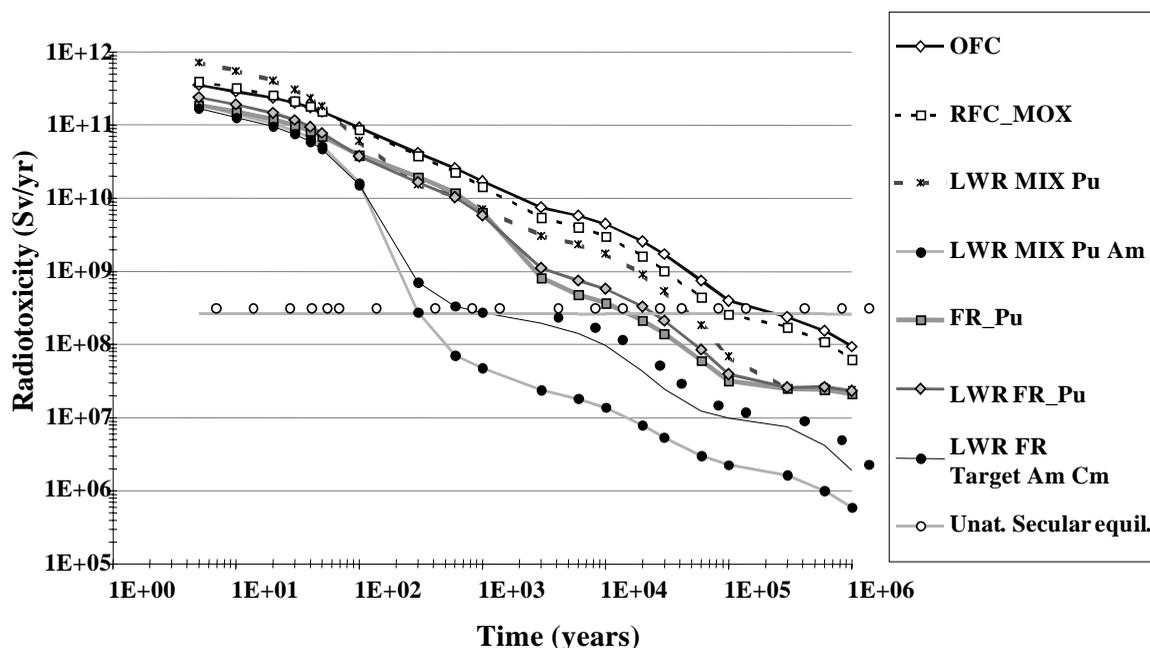




Figure 5.14 shows the potential reduction of the radiotoxicity of waste for the scenarios considered relative to the radiotoxicity of waste for the once-through LWR cycle (OFC/LWR). As a bench-mark, the radiotoxicity of natural uranium is shown also. As may be seen, the radiotoxicity of wastes going to final disposal can be reduced significantly by the advanced fuel cycles, achieving reduction factors of up to 200-300 relative to the OFC/LWR cycle. Furthermore, after significant decay times, the radiotoxicity of the advanced fuel cycle wastes are lower even than that of natural uranium. However, such a gain is based on theoretical evaluations assuming very high technical performance of the processes and plants, which remain to be proven. Thus, achievement of the indicated benefits will require long term development and decades of use. In addition, it should be remarked that the size, and to a large degree the costs, of a geological disposal site are primarily defined by the total mass of wastes, and thus mainly by the short-lived fission products rather than by the long-lived radio-nuclides. Therefore, while such advanced fuel cycle schemes are interesting from a perspective of better use of the scarce resources while reducing the potential long-term impacts, the benefits of, for example, partitioning and transmutations needs to be confronted with the socio-political criteria.

Figure 5.14 Radiotoxicity of wastes from different fuel cycles in comparison with that of natural uranium (per TWhe)



#### 5.4.2 Thorium fuel cycle

The use of thorium as a fertile material in nuclear fuel has been of interest since the dawn of nuclear power technology because of its important neutronic advantages. The absorption of a neutron in  $^{232}\text{Th}$  produces  $^{233}\text{U}$ , which is the most fissile isotope in thermal and epithermal neutron spectra. Therefore, the in-core fissile material ( $^{233}\text{U}$ ) generation capability of  $^{232}\text{Th}$  during long-term irradiation can be higher than that of  $^{238}\text{U}$  (which produces  $^{239}\text{Pu}$ ). Furthermore, in a thermal neutron spectrum the conversion ratio with thorium is better than with  $^{238}\text{U}$  because of the higher thermal neutron capture cross section of thorium (about three times higher than  $^{238}\text{U}$ ). This will reduce the initial fissile enrichment needed to achieve a specific level of burn-up.

In principle, the fuel cost and the amount of spent fuel per unit of energy generation could be reduced through the use of thorium fuel. This potential, as well as the enhanced proliferation-resistant character of the spent fuel (significantly reduced plutonium production), has led to a recent resurgence of interest in thorium fuel. As noted above, however,  $^{233}\text{U}$  is a better fissile isotope than  $^{235}\text{U}$ , so the potential proliferation aspects related to  $^{233}\text{U}$  should be assessed.

In the 1960s and 1970s, whole-core demonstrations of thorium-uranium oxide fuels in LWRs were explored in two types of arrangements:

- Mixing thorium with highly enriched uranium oxide in a uniform lattice.
- Using a heterogeneous arrangement of seed and blanket regions, in which the seed has a relatively high fissile enrichment and the blanket has a higher fraction of thorium and is responsible for most of the in-core fissile generation.

The Th/U fuel cycle was revived recently through the introduction of the Radkowsky thorium reactor (RTR) concept with a once-through fuel cycle [108]. It uses a core made of multiple seed and blanket bundles that are equal in number and outside dimensions to standard LWR uranium fuel bundles. Non-proliferation, environmental and economic benefits are claimed to be achievable through the use of uranium at medium (<20%  $^{235}\text{U}$ ) levels of enrichment, elimination of the soluble neutron-absorber systems, and simplification of the reactor control and waste management needs. Besides making use of <20% uranium enrichments in the seed and blanket, the design calls for the use of metallic uranium-zirconium fuel in the seed and use of  $\text{ThO}_2$  (or depleted  $\text{UO}_2$ ) in the blanket rods. There are two separate fuel management routes, one for the seed units and one for the blanket units. Both seed and blanket will reach >100 GWd/tHM burn-up, which reduces by about 40% the total volume of the annually discarded fuel. Plutonium production rates are estimated to be reduced by a factor of five from the current PWR rate. Reduction of plutonium and other actinide production decreases long-term radioactivity in the spent fuel, after 200 years of decay, by about 75%.

Another look at the mixed thorium-uranium dioxide once-through fuel cycle for LWRs was taken recently by INEEL [109]. Similar studies were carried out in the USA in the 1970s. Initial  $^{235}\text{U}$  enrichment is limited to 20%, as in the RTR, with 25% and 35% uranium content in the mixed thorium/uranium oxide. The economic comparison showed that the fuel costs for a  $\text{ThO}_2$ - $\text{UO}_2$  core and for an all uranium core burned to 72 GWd/tHM are about the same and about 10% higher than for an all-uranium core burned to only 45 GWd/tHM (the price of uranium and thorium were taken as the current market prices). The impact of higher (doubling) uranium prices and a somewhat reduced Th-price (USD 60/kg) result in a cost advantage of about 9% for the  $\text{ThO}_2$ - $\text{UO}_2$  case. Thorium is now co-produced with the lanthanide metals for only a few speciality uses and sells for about USD 88.5 per kilogram. The addition of commercial sources would provide for many more years of supply without upward price pressure.

Other advantages that could be claimed are related to a superior waste form, as  $\text{ThO}_2$  is the highest oxide of thorium, while  $\text{UO}_2$  can be oxidised further to  $\text{U}_3\text{O}_8$  or  $\text{UO}_3$ .

Table 5.4 shows the plutonium production and the spontaneous neutron and heat production for  $\text{UO}_2$  and Th/U fuel as reported in [109]. Pu production is found to be about 3 times less than for the pure  $\text{UO}_2$  case, with a much higher content of  $^{238}\text{Pu}$ , thus leading to a 3.2-fold greater specific decay-heat production than the conventional fuel and 40-fold higher heat than weapons grade Pu. The contents of minor actinides in the fuel is shown in Figure 5.15, in comparison with UOX and MOX fuels.

Figure 5.15 Comparison of actinide content (g/tHM) in spent fuel for UO<sub>2</sub>, MOX and Th/U fuel

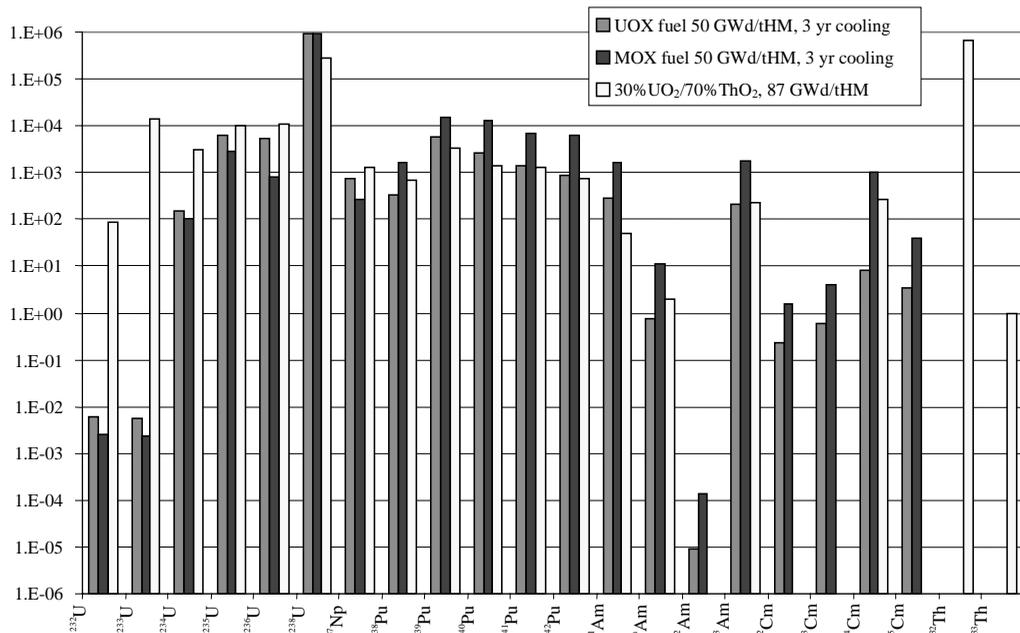


Table 5.4 Plutonium production in UO<sub>2</sub> and mixed ThO<sub>2</sub>-UO<sub>2</sub> cycles (at discharge from reactor)

|  | UO <sub>2</sub>      |             | UO <sub>2</sub>      |             | ThO <sub>2</sub> -UO <sub>2</sub> |             | ThO <sub>2</sub> -UO <sub>2</sub> |             |
|--|----------------------|-------------|----------------------|-------------|-----------------------------------|-------------|-----------------------------------|-------------|
| Irradiation time (years)                                     | 4.5                  |             | 6                    |             | 6                                 |             | 7.5                               |             |
| Burn-up (GWd/tHM)  | 45                   |             | 72                   |             | 72                                |             | 87                                |             |
| <b>Production</b>  | <b>g/kgHM</b>        |             |                      |             |                                   |             |                                   |             |
| <sup>238</sup> Pu  | 0.270                | 2.4%        | 0.697                | 4.8%        | 0.436                             | 7.8%        | 0.627                             | 9.2%        |
| <sup>239</sup> Pu  | 5.969                | 54.2%       | 7.306                | 50.5%       | 2.621                             | 47%         | 3.065                             | 45.0%       |
| <sup>240</sup> Pu  | 2.454                | 22.3%       | 3.147                | 21.8%       | 1.018                             | 18%         | 1.266                             | 18.6%       |
| <sup>241</sup> Pu  | 1.636                | 14.9%       | 2.241                | 15.5%       | 0.949                             | 17%         | 1.165                             | 17.1%       |
| <sup>242</sup> Pu  | 0.687                | 6.2%        | 1.063                | 7.4%        | 0.556                             | 10%         | 0.690                             | 10.1%       |
| <i>Total Pu</i>  | <i>11.016</i>        | <i>100%</i> | <i>14.454</i>        | <i>100%</i> | <i>5.58</i>                       | <i>100%</i> | <i>6.813</i>                      | <i>100%</i> |
| <b>Production per MWd</b>                                    |                      |             |                      |             |                                   |             |                                   |             |
| Grams Pu/MWd   | 0.245                |             | 0.201                |             | 0.078                             |             | 0.078                             |             |
| Relative   | 1.00                 |             | 0.82                 |             | 0.32                              |             | 0.32                              |             |
| Grams <sup>239</sup> Pu/MWd                                  | 0.133                |             | 0.101                |             | 0.036                             |             | 0.035                             |             |
| Relative   | 1.00                 |             | 0.76                 |             | 0.27                              |             | 0.26                              |             |
| <b>Comparison of spontaneous neutron and heat production</b> |                      |             |                      |             |                                   |             |                                   |             |
| <b>Spontaneous neutron production</b>                        |                      |             |                      |             |                                   |             |                                   |             |
| N/kg.s   | 3.72 10 <sup>5</sup> |             | 4.48 10 <sup>5</sup> |             | 5.39 10 <sup>5</sup>              |             | 5.81 10 <sup>5</sup>              |             |
| Relative   | 1.00                 |             | 1.20                 |             | 1.45                              |             | 1.56                              |             |
| <b>Decay heat</b>  |                      |             |                      |             |                                   |             |                                   |             |
| W/kg   | 16.9                 |             | 30.1                 |             | 46.6                              |             | 54.4                              |             |
| relative   | 1.00                 |             | 1.78                 |             | 2.76                              |             | 3.22                              |             |

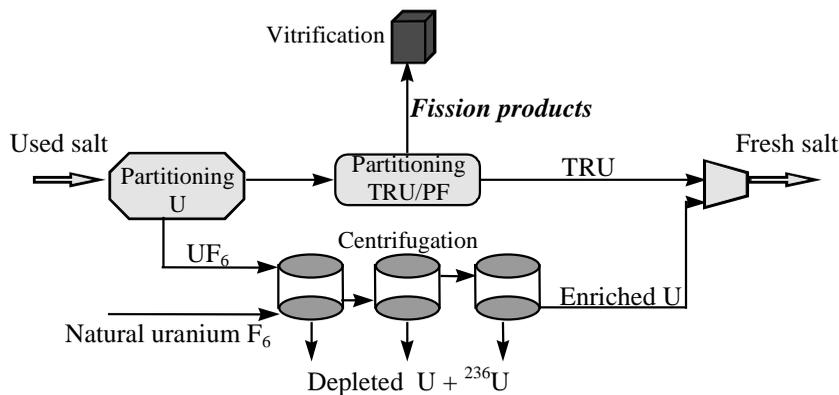
### 5.4.3. Molten salt fuel cycles

Molten salt reactors were extensively studied in the 60s and 70s. Research was carried out by the Oak Ridge National Laboratory, where an 8 MW<sub>th</sub> prototype was operated. The aim at the time was breeder reactors. Today, this type of reactor and its associated fuel cycle is again of interest to the specialists, owing to its potential as a TRU-burner.

One of the renewed concepts is studied in France by EdF and entitled the AMSTER-concept [92]. This concept uses a graphite-moderated molten salt reactor, in which the salt treatment system has been redesigned in order to reduce the waste production. Using this concept, one can define a large number of configurations according to the products loaded and recycled. Both <sup>238</sup>U or <sup>232</sup>Th support, enriched with <sup>235</sup>U if necessary, can be used in the concept.

The different fuel configurations can make use of any combination of uranium or thorium fuel and the transuranium isotopes and the long-lived fission products are multi-recycled. Figure 5.16 shows the general recycling principle.

Figure 5.16 AMSTER general recycling principle



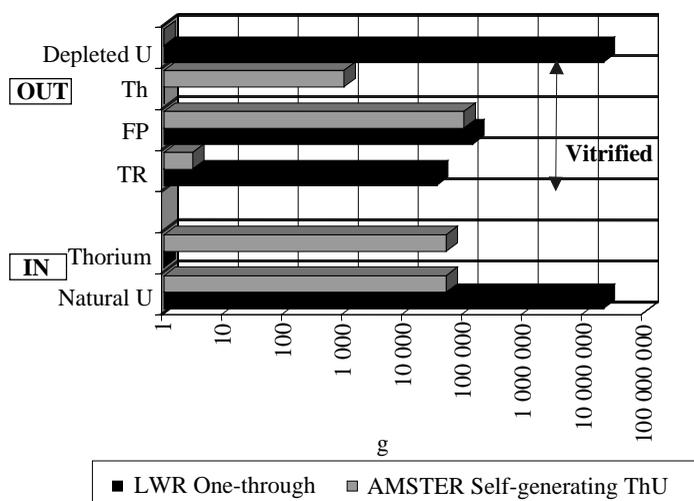
The transuranium elements are confined in the core/reprocessing unit assembly by separate extraction of the transuranium elements and rare earths by a Bismuth counter-flow system. The salt processing unit includes the cycle front-end (salt enrichment) and back-end (FP extraction).

- Cycle front-end: uranium is first extracted from the salt in UF<sub>6</sub> form, and then mixed with an appropriate mass of natural uranium (also in UF<sub>6</sub> form). This mixture is enriched with <sup>235</sup>U if necessary, to reach the required <sup>235</sup>U enrichment value in the new salt, for example using an ultra-centrifuge. This solution would require a small number of centrifuges, owing to the small quantity of <sup>235</sup>U to be added. The residual depleted uranium is evacuated, taking with it a large proportion (about 35%) of the <sup>236</sup>U in the spent salt. This prevents the core from becoming poisoned by build-up of <sup>236</sup>U.
- Cycle back-end: the transuranium elements are separated in a salt-liquid metal exchanger. Given the good separation factor (about 10) in this operation, and by using 6 consecutive stages, the salt would contain only a very small residue (about 10<sup>-5</sup>) of the initial mass of transuranium elements. The thorium and the fission products, except for the long-lived FP to be incinerated, then are extracted from the salt with no need for a high separation capacity. The residual transuranium elements in the salt are extracted with the FP, which can be vitrified and stored in the same manner as the fission products.

In the case of a thorium support, the consumption of fissile uranium is low enough for it to be produced in the form of  $^{233}\text{U}$ , in an extra core zone. This fertile zone, located on the core periphery, would be under-moderated by increasing the diameter of the salt holes. In this concept, the size of the fertile zone would be adapted to make the reactor only just a  $^{233}\text{U}$  self-generator (the production of the fertile zone would exactly compensate the consumption of the fissile zone). Salt processing would be by simple removal of the FP and replacement by the same mass of  $^{232}\text{Th}$  and/or  $^{238}\text{U}$ .

The AMSTER concept as a molten salt reactor fuel cycle system should allow the production of energy with very reduced quantities of transuranium element waste (a few g per TWhe), and with no transportation of highly radio-toxic substances. A mixed thorium-uranium support with a peripheral fertile zone, breeding its own uranium, should further improve this performance. This fuel scheme should in fact offer incinerating performance identical to that of a thorium reactor supplied with  $^{235}\text{U}$ , while eliminating the need for uranium enrichment and making the reactor non-proliferating. Furthermore, this self-breeding reactor would consume only 50 kg of thorium and 50 kg of  $^{238}\text{U}$  per TWhe which, in the light of estimated available resources, would allow the production of 20 to 70 million TWhe. The depleted uranium stored in France (200 000 t), together with 200 000 t of thorium, could produce 4 million TWhe (current annual nuclear electricity production is 2 540 TWhe). Figure 5.17 compares the mass balance entering and leaving the reactor for a standard open-cycle PWR and a self-generating AMSTER with a support of 50% U and 50% Th. The inputs are natural or depleted uranium and thorium. The outputs comprise Transuranium elements, U and Th all of which are vitrified with the FP, and depleted uranium produced from enrichment.

Figure 5.17 Mass balance entering and leaving the reactor for a standard open-cycle PWR and for a self-generating AMSTER with 50% U and 50% thorium



## 5.5. Conclusions

The previous sections highlighted the wealth of developments currently under consideration and showed that, although most of the long-term systems still remain conceptual or currently are progressing to the laboratory/project phase, the advanced systems are significantly more sustainable through the reduced demand for resources and more environmentally friendly through the reduced production of waste. These ongoing developments currently have not considered the economic and social dimensions in great detail, since the early status of most of these concepts mean that the R&D efforts remain related more to the technical aspects than to the economic and socio-political aspects.

## **6. CONCLUSIONS AND RECOMMENDATIONS**

### **6.1 Introduction**

This study had the overall objective of providing an overview of the currently perceived (from the viewpoint of a group of experts) main trends for future development of the nuclear fuel cycle. These trends were examined in terms of their potential for continuously improving the performance of the nuclear fuel cycle, such as better resource utilisation, higher technical performance, lower costs, and enhanced responsiveness to socio-political issues and concerns.

The Expert Group recognised that nuclear fuel cycle development is an on-going process, that will be carried out as long as nuclear energy continues to be utilised, and that it is not possible to foresee all developments that might be possible in the very long term. Therefore, in order to make its task more manageable within resource and time limits for the study, the Expert Group focused its attention on those near-term (next 25 years) and mid-term (25-50 years into the future) trends that appear to be most relevant in ensuring that nuclear energy can be maintained as a safe, economic and sustainable component of the world's energy supply system.

The previous chapters introduced some specific conclusions related to technical developments in the nuclear fuel cycle. Building on those chapters, this chapter presents an overall synthesis of the observed development trends, and provides some overall conclusions drawn by the experts.

### **6.2 The nuclear fuel cycle and sustainable development**

Nuclear energy has some special characteristics, which differ significantly from other sources of energy, with respect to sustainable development. Indeed, it is exactly those special characteristics that are at the heart of a number of the arguments for, as well as those against, nuclear energy.

From one viewpoint, many scientists and experts, and part of the media and public, see nuclear energy as being uniquely capable of providing long-term supplies of clean, safe and economic energy in the large quantities that will be needed to support a growing global population and to permit continued economic development world-wide. In support of this view, they point to special characteristics for nuclear energy such as: being essentially free from emissions of greenhouse gases (GHG); having a large natural resource base (uranium and thorium) with no competing demands; offering the potential to be a non-exhaustible source of energy, through the introduction of breeder reactors and advanced fuel cycles, and eventually the extraction of uranium from seawater.

From another viewpoint, diametrically opposed to the above, critics of nuclear energy point to a different set of special characteristics, related to: risks of nuclear accidents; problems in the long-term management of spent nuclear fuel and radioactive wastes; and threats of nuclear weapons proliferation. They argue that these issues are of such over-riding importance, and so unlikely to be addressed successfully, as to make nuclear energy “not sustainable” as a long-term component of the world's energy system.

All the development trends in the nuclear fuel cycle are aiming toward the target of improving, by one way or another, the economic, environmental and social aspects of nuclear energy. Successfully meeting this target would help ensure that the first viewpoint cited above continues to be supported, and also could go a long way towards solving the perceived problems that are put forth by those holding the second viewpoint, thereby contributing to a broader acceptance of an expanded role for nuclear energy in the future.

### **6.2.1 Resource sustainability**

Although there are some uncertainties regarding the exact quantities of uranium and thorium resources on earth, it is concluded that they are sufficient, when used in breeder reactors with advanced fuel cycles, to enable nuclear energy to be a very long-term or almost inexhaustible source of energy. The world's "conventional" resources of uranium are known to be adequate to support the current level of nuclear energy production for the coming 80 years or so, using current nuclear power plant and fuel cycle technologies. This compares well with the time-period that can be supported by the estimated "conventional" reserves of oil and gas. Furthermore, the effectiveness of using uranium resources can, in principle, be increased by almost a hundred-fold by the use of fast breeder reactors together with the recycling of plutonium and uranium. Therefore, development and large-scale implementation of advanced nuclear power and fuel cycle technologies would permit uranium resources to meet requirements for hundreds, if not thousands, of years.

Lastly, extraction of uranium from "unconventional" sources, notably from seawater, potentially could provide an additional vast energy resource, that would extend even further the time period for which nuclear energy production could be sustained, if future technological developments can make this option economically viable.

Thorium, which is three times more abundant in the earth's crust than uranium, represents an additional primary resource for nuclear energy production. The development and implementation of reactors and fuel cycles for exploiting the thorium resources could make yet another large contribution to the long-term sustainability of the nuclear energy option.

While it is recognised that nuclear fission might be replaced by even more sustainable energy sources, e.g. nuclear fusion, in order to meet the world's ever increasing demand for power, the time-period necessary to deploy these energy sources could extend well beyond the term considered for this study, even if R&D efforts would be fully successful. For the next fifty years, nuclear fission will need to continue to fulfil a major role as an important component of a diversified energy mix.

### **6.2.2 Economics**

Nuclear power has a very high degree of long-term stability with respect to possible increases in the price of the nuclear fuel raw material (uranium). The cost of natural uranium is about 25% of the total nuclear fuel cycle cost and only some 5% of the total nuclear electricity generating cost, whereas fuel costs represent up to 80% of fossil fuel electricity generation cost. Thus, a doubling of uranium price (not considered very likely in the short term) would increase the nuclear generating cost by only 5%, whereas a doubling of natural gas prices (which has been experienced during the last 2-3 years) would increase gas-fired generating costs by some 80%. In addition, the wide geographical distribution of uranium resources, ease and low cost of maintaining stockpiles, and relatively low cost of the fuel cycle operations, make it unlikely that possible changes in prices anywhere in the nuclear fuel cycle would reduce significantly the economic competitiveness of nuclear relative to other forms of energy in the foreseeable future.

It is of particular importance to note that provisions for waste management and disposal, as well as for the decommissioning of the power plants, already are “internalised” in the costs of nuclear electricity production, and thereby included in the prices paid by consumers of nuclear generated electricity. This generalised practice in the nuclear industry of the OECD countries is in contrast with classical industries where the consequences of emissions (e.g. greenhouse gases, dust particles) are left to be borne by the population at large. Any comparison of the costs and benefits of alternative energy sources should reflect such variations in practices for accounting for the total costs to society of different means of energy production.

The higher capital investment cost of nuclear power plants requires longer financial commitments than for other power plant types, but the lower and more stable cost of the nuclear fuel cycle results in a more predictable energy cost, once the power plants are built. This favourable competitive position of nuclear energy is expected to be enhanced by the anticipated future improvements to the nuclear fuel cycle.

### **6.2.3 Environment**

As reported in several independent reviews by international organisations, national scientific academies or expert groups, as well as surveys of public opinion, nuclear energy is considered to be the only technologically proven large-scale energy source that can supply a growing world population with energy that is free from emissions of gases or particles that contribute to acid rain, urban smog, depletion of the ozone layer or climate change. With regard to the latter in particular, carbon dioxide emissions are negligible throughout the entire nuclear fuel cycle, and nuclear energy already avoids some 8% of CO<sub>2</sub> that would be emitted world-wide if the current nuclear electricity generation would be produced instead by fossil fuels.

The quantities of wastes arising from nuclear energy are small in comparison with the wastes of other major energy sources, and this makes it possible to collect and isolate them in ways that ensure acceptably low impacts to human health and the environment, and at acceptable costs. Technical solutions exist for the ultimate disposal of these wastes though some of the solutions, notably deep geological disposal of high-level waste, still are awaiting socio-political agreement prior to being fully implemented. With regard to the latter, it is of high importance that the nuclear industry be able to demonstrate, through pilot or full-scale projects, the practicality of these technical solutions for final disposal of its high level waste, in order to enhance public support.

There are, however, ways to further reduce the long-term impact, albeit already small and controllable, of minor actinides present in the spent fuel. These ways call for advanced fuel cycles. Fast reactors and/or thorium based fuel cycles seem to offer potential in the medium to long-term with respect to applicability and economics.

Much of the current public opposition to nuclear energy is focused on the transport of spent fuel and high level waste, in spite of the fact that the industry has accumulated more than 40 years of experience in spent fuel and high level waste transport, without experiencing a single accident with radiological consequences to the public or the environment.

### **6.2.4 Social**

During its more than 50 year history, only a small number of accidents with significant radiological consequences have occurred in the OECD Member countries in a relatively few facilities of the nuclear fuel cycle. Most of these accidents or incidents related to criticality accidents, releases of UF<sub>6</sub>, fire and exothermic reactions, losses of or contamination by radioactive material, and losses of

electric power in fuel cycle facilities. The number of fatalities to workers in the facilities amounted to about 10 while the public at large did not experience any harmful irradiation or contamination [57]. These events, while rare, call for a strict compliance with the regulatory requirements and quality management in every stage of the fuel cycle industry (i.e. a high level of “safety culture”).

Routine radiological impacts of the nuclear fuel cycle on the environment are very limited, to levels well below regulatory limits (which are set at levels sufficiently low as to ensure no effects on humans or the environment), and represent only a very minor fraction of the radiation received by the public from natural or other artificial sources (e.g. medical X-rays) of radiation. Radiological impacts on workers are low and strictly controlled. The industry must, however, remain vigilant against accidents or incidents, taking into account that, as noted above, these have occurred occasionally, although there have been no significant consequences to the public.

Proliferation risk has also been part of public and political concerns. However, proliferation risks are judged differently by different experts. Some do see proliferation as a key issue, while others make reference to the fact that there has been “no-case-of-experience” over the last decades and reject proliferation concerns as a misperceived artificial problem. In the view of most experts, the institutional safeguards regime that has been established (e.g. the IAEA inspection system) is adequate to ensure that there will be no undetected diversion of material from the civil nuclear fuel cycle for military or terrorist purposes. Nonetheless, new developments in reactor and nuclear fuel cycle technologies are also enhancing the “proliferation resistance” of fuel cycle facilities, in order to provide even higher assurance that such diversion will not occur. In this regard, the INFCE study [43] indicated that “integration of reprocessing in the reactor plant is advantageous” and “increased build-up of higher isotopes through multiple recycling makes plutonium less attractive for theft”.

### **6.3 Challenges for future development**

Given today’s market-driven environment, there is limited potential for industry to fund the long-term R&D needed to develop and deploy the advanced fuel cycles discussed in this report. Political pressures and competing budget priorities have worked to reduce nuclear R&D funding by Governments as well. While there are some signs that Government funding may increase in the near future, funding constraints are likely to limit the number of the discussed fuel cycle options that can be developed.

Comprehensive planning that includes consideration of economic, environmental and social factors in a well-balanced and integrated comparative assessment of different options will become increasingly important in formulating and taking decisions on long-term energy policies, including for nuclear power. Such comprehensive planning in the nuclear sector will be useful also in selecting, as highest priority for development, those nuclear power plant and fuel cycle alternatives that offer the most favourable sustainability characteristics and that build best on existing technologies [110,111].

As described in this report, a range of developments is going on, or are projected, to further develop nuclear energy. While nuclear energy already has a very high level of safety and is economic in a number of countries, continuing activities are aiming towards further deploying the potential of nuclear power and its acceptance by the public.

Nonetheless, nuclear power faces major challenges that must be overcome, for example, in the following areas:

- Deployment of advanced reactor concepts and fuel cycles will remain a lengthy and expensive process. Multilateral or international co-operative R&D programmes will therefore become increasingly important in order to pool limited financial resources and obtain the benefits of synergy among R&D activities, thereby shortening the process of proceeding from concept to industrial reality. Indeed, there already are some examples of such co-operative activities, including the USA-initiated Generation IV International Forum (GIF), the IAEA-led INPRO, and the multi-national interest in the PBMR.
- Facilities for the ultimate disposal of spent fuel and high-level wastes should be implemented, in order to demonstrate to the public that the industry is responsible in managing its wastes, and that the disposal systems, for which the necessary technologies already exist, can be operated with very limited impact on the environment. The stakeholders may also consider it important to further develop advanced reactors and fuel cycles, in order to reduce the amount of wastes for disposal or to reduce their required confinement times, as well as to improve the efficiency of using of natural nuclear fuel resources.
- Techniques for uranium resource conservation or extension of these resources by Pu-utilisation/recycling (including ex-military Pu) are to be further developed. The most effective resource utilisation can be reached only by use of advanced reactor systems (i.e. fast breeder reactors or thorium fuelled reactors).
- The nuclear fuel cycle may also help to minimise the potential for the diversion of nuclear materials for weapons purposes, i.e. by disposition of ex-weapons material, reducing the Pu-inventory and others.

## 6.4 Conclusions

- When analysed against a large set of criteria, nuclear power shows today a unique potential as a large-scale sustainable energy source. New reactor types and their associated advanced fuel cycles could extend greatly the time frame of its sustainability and allow a greatly expanded use, as well as opening nuclear applications beyond today's electricity production (e.g. for process heat, hydrogen production), which could significantly expand the role that nuclear energy could play in a sustainable development future.
- The nuclear fuel cycle is an important element in the overall acceptability of nuclear energy, but there is nevertheless potential for further improvements. To obtain these improvements, short-term ongoing industrial development programmes in different stages of the nuclear fuel cycle (enrichment, fuel performance, reprocessing techniques, waste disposal concepts) comprise important areas for further cost-reductions and environmental improvements.
- There is a global market in the front end of the fuel cycle. Existing facilities may be adequate to face the present demand, but there will be a problem of replacement and development of these facilities beyond this decade, especially if nuclear power is to pick up momentum.



## *Annex A*

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## *Annex B*

### FUEL CYCLE CALCULATIONS

The calculations presented in Chapter 5 have been made mainly by CEA, using the COSI code (described in detail in [a]). This code is able to simulate the mass fluxes of various radionuclides, through the various steps of a specific reactor park. JNC has given the data for the mixed LWR-FR without MA recycling cycle, and for the pure FR park with MA recycling (not presented in the study).

Here, steady state calculations have been carried out to make cycle comparisons relevant. No results are given concerning the transient periods, starting from today's situation in a given country and progressively leading to the steady states described here. This has been made in the case of the French situation [b]. It is to be noted that the time it takes to reach steady state can be very long in some cases (up to one hundred years or more).

More details concerning the French calculations are given below.

Steady state equilibria calculated with COSI were established for PWRs through modelling based on APOLLO2 [c] and for FRs through modelling based on the ERANOS code [d]. Constraints imposed by fuel cycle material management were not taken into consideration, specifically the transition, warehousing management and the reactor unit commissioning aspects.

The efficiency of the reprocessing is an important parameter for calculating the amount of final waste in the cases where reprocessing is needed (virtually all the scenarios except the OFC and CANDU).

The following table indicates the values used for calculating the mass fluxes of the various elements in the reprocessing plant.

| Element             | Gaseous waste % | Liquid waste % | Hulls % | I traps % | Glass packages % | Glass packages advanced % | Total % |
|---------------------|-----------------|----------------|---------|-----------|------------------|---------------------------|---------|
| <sup>3</sup> H      | 0.5             | 39.5           | 60.0    |           |                  |                           | 100.0   |
| C                   | 10.0            | 90.0           | 0       |           | 0.0              | 0.0                       | 100.0   |
| activation products |                 |                | 100     |           | 0.0              | 0.0                       | 100.0   |
| gaseous FPs         | 99.8            |                | 0.2     |           |                  |                           | 100.0   |
| solid FPs           |                 | 0              | 0.2     |           | 99.8             | 99.8                      | 100.0   |
| I                   | 1               | 98.1           | 0.2     | 1         |                  |                           | 100.0   |
| U                   |                 |                | 0.050   |           | 0.050            | 0.050                     | 0.1     |
| Pu                  |                 |                | 0.050   |           | 0.050            | 0.050                     | 0.1     |
| Np                  |                 |                | 0.050   |           | 99.950           | 0.050                     | 100.0   |
| Am                  |                 |                | 0.050   |           | 99.950           | 0.050                     | 100.0   |
| Cm                  |                 |                | 0.050   |           | 99.950           | 0.050                     | 100.0   |

The MA content of the hulls and fuel element components has been assumed to be 0.05% of the MA content of the spent fuel. This assumption is considered too conservative, but the measurements are very difficult and the results not definitive today. Measured figures range from 0.1% to 0.004%, depending on the measurement and the element considered. We have taken an intermediate value. This has an impact on the final toxicity of the ultimate waste.

The overall extraction efficiency of U and Pu has been assumed to be 99,9%, according to the present industrial performance in La Hague, France. The same number has been assumed to be valid in the case of advanced reprocessing for MAs, in accordance with the objectives of the scientific programme in France for advanced separation.

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## LIST OF ACRONYMS

|               |   |
|---------------|---|
| ABB           | Asea Brown Boveri (Sweden)  |
| ABWR          | Advanced Boiling Water Reactor  |
| ADS           | Accelerator Driven Systems  |
| AECL          | Atomic Energy of Canada Limited (Canada)  |
| ANL           | Argonne National Laboratory (USA)   |
| APA           | Advanced Plutonium Assembly   |
| APWR          | Advanced Pressurised Water Reactor  |
| ATR           | Advanced Thermal Reactor (Japan)  |
| ATW           | Accelerator Treatment of Waste  |
| AVLIS         | Atomic Vapour Laser Isotope Separation (SILVA in France)  |
| AVR           | “Arbeitsgemeinschaft Versuchsreaktor GmbH”<br>(pebble-bed high temperature gas-cooled reactor – see also PBR)   |
| BN            | Belgonucléaire  |
| BNFL          | British Nuclear Fuel plc. (UK)  |
| Bq            | Becquerel   |
| BU            | Burn-up   |
| BWR           | Boiling Water Reactor   |
| CAMECO        | Canadian Uranium Mining Company   |
| CANDU         | Canadian Deuterium Uranium Reactor (see also PHWR)  |
| CEA           | Centre d’étude atomique (France)  |
| CERCER        | Ceramic ceramic fuel form   |
| CERMET        | Ceramic metal fuel form   |
| CNNC          | China National Nuclear Centre (China)   |
| COCA          | Cobroyage Cadarache (France)  |
| COGEMA        | Compagnie générale des matières nucléaires  |
| CRIEPI        | Central Research Institute of Electric Power Industry (Japan)   |
| DECADES       | Database and Methodologies for Comparative Assessment of Different<br>Energy Sources for Electricity Generation |
| DOE           | Department of Energy (USA)  |
| DSM           | Demand Side Management  |
| DUPIC         | Direct Use of PWR spent Fuel in CANDU   |
| EAR           | Estimated Additional Resources  |
| EC            | European Commission   |
| EIA           | Environmental Impact Assessment   |
| ENUSA         | Enusa Industrias Avanzadas, S.A.  |
| FBR           | Fast Breeder Reactor (see also FR, LMFR, LMFBR)   |
| FENCH         | Full Energy Chain   |
| FP            | Fission Product   |
| FR            | Fast Reactor (see also FBR, LMFR, LMFBR)  |
| FRAMATOME ANP | Merger of nuclear activities of Framatome and Siemens GmbH  |
| GBq           | Giga Becquerel (10 <sup>9</sup> Becquerels)   |
| GCR           | Gas-cooled Reactor  |

|         |   |
|---------|---|
| GDR     | German Democratic Republic  |
| GE      | General Electric (USA)  |
| GHG     | Greenhouse Gases  |
| GT-MHR  | Gas Turbine – Modular Helium Reactor (see also MHR, MHTGR)              |
| GWd     | Gigawatt days   |
| GWe     | Gigawatt electric   |
| GWP     | Global Warming Potential  |
| HEU     | Highly Enriched Uranium (see also LEU)                                  |
| HMR     | High Moderation Reactor   |
| HTGR    | High-temperature Gas-cooled Reactor (see also GT-MHR, HTR, MHR)         |
| HTR     | High-temperature Reactor (see also GT-MHR, HTGR, MHR)                   |
| HTRR    | High-temperature Test Reactor   |
| IAEA    | International Atomic Energy Agency                                      |
| IEA     | International Energy Agency (OECD)                                      |
| IFR     | Integral Fast Reactor   |
| ILW     | Intermediate Level Waste  |
| IMF     | Inert Matrix Fuel   |
| ISL     | In Situ Leaching  |
| JAERI   | Japan Atomic Energy Research Institute                                  |
| JNC     | Japan Nuclear Cycle Development Institute                               |
| JNFL    | Japan Nuclear Fuel Limited (Japan)                                      |
| KAERI   | Korea Atomic Energy Research Institute (Korea)                          |
| KNFC    | Kepeco Nuclear Fuel Co., Ltd.   |
| KNGR    | Korea Next Generation Reactor   |
| kWh     | Kilowatt-hour   |
| LANL    | Los Alamos National Laboratory (USA)                                    |
| LCA     | Life Cycle Analysis   |
| LEU     | Low-enriched Uranium (see also HEU, SEU)                                |
| LINEX   | Lithium Nitrate Extraction of Actinides                                 |
| LLW     | Low-level Waste   |
| LMFR    | Liquid Metal Fast Reactor   |
| LMFBR   | Liquid Metal Fast Breeder Reactor (see also FR, FBR)                    |
| LWR     | Light-water Reactor   |
| MCA     | Multi-Criteria Analysis   |
| MA      | Minor Actinides   |
| MHR     | Modular Helium Reactor (see also GT-MHR, MHTGR)                         |
| MHTGR   | Modular High-temperature Gas-cooled Reactor (see also GT-MHR, HTR, MHR) |
| manSv   | Man-Sievert   |
| MIMAS   | Micronized Masterblend  |
| MINATOM | Russia's Atomic Energy Ministry   |
| MNF     | Mitsubishi Nuclear Fuel Co., Ltd (Japan)                                |
| MOX     | Mixed (uranium + plutonium) Oxide                                       |
| MPC     | Maximum Permissible Concentration                                       |
| MSR     | Molten Salt Reactor   |
| NEA     | Nuclear Energy Agency (OECD)  |

|                  |  |
|------------------|--|
| NERI             | Nuclear Energy Research Initiative (USA)   |
| NFI              | Nuclear Fuel Industries, Ltd. (Japan)  |
| NIMBY            | Not In My Back Yard  |
| NPP              | Nuclear Power Plant  |
| O & M            | Operation and Maintenance  |
| OECD             | Organisation for Economic Co-operation and Development   |
| OSPAR            | Oslo-Paris Convention for the Protection of the Marine Environment of the North-East Atlantic  |
| OREOX            | Oxidation and Reduction of Oxide Fuels   |
| P&T              | Partitioning and Transmutation (of fission product isotopes)   |
| PBMR             | Pebble Bed Modular Reactor   |
| PCI              | Pellet-cladding Interaction  |
| PHWR             | Pressurised Heavy Water Reactor (see also CANDU)   |
| PSI              | Paul Scherrer Institute (Switzerland)  |
| Pu               | Plutonium  |
| PuO <sub>2</sub> | Plutonium Dioxide  |
| PUREX            | Plutonium Uranium Extraction   |
| PWR              | Pressurised Water Reactor  |
| RAR              | Reasonably Assured Resources   |
| R&D              | Research & Development   |
| RE               | Rare Earths  |
| REPU             | Reprocessed Uranium  |
| RIAR             | Research Institute of Atomic Reactors (Russia)   |
| ROX              | Rock-like Oxide  |
| SEU              | Slightly-enriched Uranium (see also LEU)   |
| SILEX            | Separation of Isotopes by Laser Excitation   |
| SKB              | Svensk Kärnbränslehantering AB   |
| SWU              | Separative Work Unit   |
| TBP              | Tributyl Phosphate   |
| TRU              | Transuranic/Transuranium isotopes  |
| tHM              | Tonnes heavy metals  |
| TRUEX            | TRU Extraction   |
| tU               | Tonnes Uranium   |
| UNSCEAR          | United Nations Scientific Committee on the Effects of Atomic Radiation   |
| UOX              | Uranium oxide  |
| UO <sub>2</sub>  | Uranium dioxide  |
| URENCO           | Urenco plc., developed from a joint Dutch, German and British initiative set up in the 1970s following the signing of the Treaty of Almelo |
| US               | United States  |
| USA              | United States of America   |
| USEC             | United States Enrichment Corporation (USA)   |
| USSR             | Union of Soviet Socialist Republics  |
| VVER             | Russian type of Pressurised Water Reactor  |

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