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Back-end of the Fuel Cycle in a 1000 GWe Nuclear Scenario

Workshop Proceedings
Avignon, France
6-7 October 1998

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NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter; Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all OECD Member countries, except New Zealand and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

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

FOREWORD

Nuclear power is one of the carbon-free electricity generation options that can help alleviate the risk of climate change. Nuclear power continues to contribute to the lowering of carbon intensity in the energy sector. Today, greenhouse gas emissions are approximately 7% lower than they would have been in a non-nuclear scenario. The OECD Nuclear Energy Agency is thus examining the conditions that would allow nuclear energy to continue playing an important role in low carbon intensive energy supply through 2050 and beyond.

In this context, it is important to assess the technical and economic feasibility of alternative nuclear development paths as well as their sustainability. The optimisation of the nuclear fuel cycle is a key issue for the sustainability of nuclear energy. The overall objective of the workshop on "The Back-End of the Fuel Cycle in a 1 000 GWe Nuclear Scenario" was to investigate alternative options for the back-end of the fuel cycle and to assess their capability to enhance the sustainability of nuclear power in the long term, to 2050 and beyond. The workshop covered natural resource management, radioactive waste minimisation (volumes and activity), cost reduction and proliferation resistance aspects of alternative reactor and fuel cycle technologies and strategies.

These proceedings include the papers presented and discussed during the workshop. The opinions expressed are those of the authors only and do not necessarily reflect the views of OECD Member countries or international organisations represented in the meeting. This book is published under the responsibility of the Secretary-General of the OECD.

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* Paper submitted by the author to the Secretariat as a summary of his oral intervention during the workshop.

OPENING REMARKS

by Mr. Philippe Savelli
Deputy Director, Science, Computing and Development,
OECD Nuclear Energy Agency

Ladies, Gentlemen, dear Colleagues,

I am delighted and honoured to welcome you in these distinguished surroundings on behalf of the OECD Nuclear Energy Agency, and to open this workshop on the back-end of the fuel cycle in a 1 000 GWe nuclear scenario in the year 2050. First of all, I would like to extend a warm welcome on behalf of the Agency and myself to the Commissariat à l'Énergie Atomique, whose invitation gave us the opportunity to meet here. My thanks also go out to the International Atomic Energy Agency, which helped the NEA to organise this workshop and actively participated in drawing up the programme for it. Lastly, and above all, I am grateful to all of you, particularly the speakers and the Chairmen of the sessions, for being with us today in order to share your knowledge and exchange your ideas on the technical and strategic issues we must address if optimum solutions for the back-end of the nuclear fuel cycle are to be implemented.

Two committees in the Agency, the Committee for Technical and Economic Studies on Nuclear Energy and the Fuel Cycle and the Science Committee took part in the organisation of this workshop. The workshop programme reflects their co-operation, covering the numerous scientific, technical and strategic facets of the back-end of the nuclear fuel cycle.

In deciding to include this meeting in its work programme, the Agency had in mind two of the major concerns of decision-makers involved at national level in working out strategies for the peaceful use of nuclear energy: sustainable development, and the drawing up of research and development programmes suited to future needs. It is essential to implement optimised solutions for the back-end of the fuel cycle in order to make nuclear energy a lasting component of national and global energy supply in the long term. They can only be put in place if the necessary technologies are available in due course, which means that research and development programmes, either under way or planned, should be equal to the challenges we shall have to face between now and 2050.

In organising this workshop, our aim was to provide an opportunity for specialists and decision-makers to exchange views and discuss issues with a view to identifying priorities for national R&D programmes and the activities of international organisations. Moreover, the lessons drawn from our discussion will be part of the Agency's contribution to the OECD's work on sustainable development. The workshop's conclusions will be incorporated into the synthesis reports being prepared in the OECD on the role of technologies in finding ways of reducing the risks of climate change and of ensuring sustainable development.

Your presence here today is a demonstration, if such were needed, of the importance attached by the Member countries of the NEA to the future of nuclear energy and the problems raised by the back-end

of the fuel cycle. The diversity of the topics to be addressed by the speakers and discussed in the course of the next two days testifies to the scope and quality of the work being done world-wide to improve existing technologies and develop new solutions in order to meet the goals of optimising the use of natural resources, increasing the competitiveness of nuclear-generated electricity, reducing the impact on the environment, in particular minimising the volume and activity of radioactive waste and strengthening guarantees of the non-proliferation of nuclear weapons.

During this workshop we shall, of course, discuss technologies already being used on an industrial scale, such as the reprocessing of light-water reactor fuel and the management of the associated waste, as well as the recycling of plutonium in mixed oxide fuel. We shall also tackle emerging technologies – whether under study or undergoing development – in regard to both reactor concepts and the fuel cycle as a whole.

Our ambition is not to identify "the" solution, inasmuch as each country has its own assessment criteria which depend, inter alia, on the national economic context, domestic energy resources and existing or planned nuclear electricity programmes. On the other hand we ought to be able to assess the various options in the light of the criteria I have just mentioned and, where appropriate, estimate the duration and scale of the R&D required to move from the conceptual to the industrial phase and thence to the commercial development of innovative options.

I hope this workshop will allow an in-depth discussion of some of the questions raised by the work carried out recently by the NEA and the IAEA, such as the study published by the NEA at the beginning of this year on nuclear energy and climate change, which drew attention to problems of the back-end of the cycle in an ongoing growth scenario for nuclear generated power. In the addresses we are going to hear, solutions to some of these problems will be outlined. The symposium organised by the IAEA, the NEA and other international organisations on the adaptation of reactor and fuel cycle strategies to the new realities, which was held last year in Vienna, concluded amongst other things that international co-operation in the field of the fuel cycle is essential if the challenges of the next century are to be met. This workshop is part of the action being undertaken by the two agencies to strengthen such co-operation.

For the NEA, another purpose of this workshop is to guide future activity in the field of the fuel cycle in accordance with Member countries' needs. It will serve in particular to define more precisely the fuel cycle optimisation project which the NDC has included in its work programme for 1999-2000.

At a time when economic imperatives are forcing many countries to reduce their R&D expenditure, particularly that financed by government, the exchange of information seems to me essential in order to ensure that national endeavours complement each other and to increase their effectiveness. One of the roles of intergovernmental organisations like the NEA and the IAEA is to promote such exchanges and, at the request of Member countries, to initiate projects of common interest with a view to identifying topics for analysis and areas of consensus that might provide a basis for defining national strategies suited to future needs. I hope this workshop will help us fulfil this role effectively.

Before concluding, I would like to thank Bruno Sicard and his team, who were responsible for the practical organisation of this workshop. I know from experience how difficult it is to arrange an international meeting and I am sure that it took many hours of work to put in place a framework conducive to fruitful discussion.

It remains for me only to wish you a stimulating discussion, a pleasant stay in Avignon and, for those going on the visit to the Atalante laboratories, the chance to discover some of the research and development resources for the back-end of the fuel cycle that have been put in place in France.

OPENING REMARKS

by Mr. Yves Lapierre
Deputy Director, Fuel Cycle Department
Commissariat à l'Énergie Atomique

Mr. Director,

Ladies, Gentlemen, dear colleagues,

On behalf of the Commissariat à l'Énergie Atomique (CEA) and Noël Camarcat, the Fuel Cycle Director, I am delighted and honoured to open this workshop on the back-end of the fuel cycle in a 1 000 GWe nuclear scenario in the year 2050.

I find it particularly symbolic that this workshop is being held in the Rhône valley in France. As far as the nuclear sector is concerned, this place is steeped in history and also prefigures the future. We are very near Marcoule and Pierrelatte, which were the cradle of the fuel cycle industry in France, both the front-end and back-end. Pierrelatte developed the gaseous diffusion process for the enrichment of uranium and is today the site of the George Besse Eurodif plant, an essential component of France's energy independence. Marcoule is the site that was chosen to develop the back-end of the fuel cycle technology. The bulk of the research that contributed to the success of the La Hague plant was conducted there.

Today these sites are undergoing fundamental change. The facilities that made the industrial developments of the cycle possible have come to the end of their lives. The UP1 plant was shut down nearly a year ago. The Marcoule pilot plant is being cleaned up. But this end of an era also marks, like the phoenix rising from its ashes, the will to develop the technology that will provide the nuclear energy of 2050, the timeframe for this workshop.

The nuclear sector, if it is to meet the challenge of 1 000 GWe by 2050, will have to address several issues that will be crucial to its continuing existence and it will have to do this probably before then, i.e. by 2010 or 2020. We have to improve the competitiveness of both the front-end and back-end of the fuel cycle; to ensure that light water reactor systems are renewed and identify the new fuels; and to encourage acceptance of nuclear energy by the public. This means improving waste management still further, confirming the feasibility of dismantling facilities, etc.

In France, the Commissariat à l'Énergie Atomique, together with its industrial research partners, plays a major role in solving these problems. The Fuel Cycle Department of the CEA is responsible for most of the issues referred to: the front-end and back-end of the cycle in co-operation with Cogéma, waste management in co-operation with all those working in the nuclear sector (EDF, Cogéma, Andra), cleaning up and dismantling the CEA's nuclear installations.

During this workshop, I am sure that these issues will be addressed.

Monsieur Savelli mentioned several points that seem to me essential. Our discussions should enable us to find credible solutions. These solutions cannot be local ones and must answer the questions each of us is asking, bearing in mind the specific features of our economic, social or international context. The answer can only be a collective one.

International co-operation can play a major role in helping us to solve our problems – international co-operation in research and probably industrial co-operation as well. That is why I would like us to take advantage of the opportunity offered by the NEA to share our questions and our tentative solutions.

We are going to try to make a projection up to the year 2050. Such a projection is meaningful only if the nuclear sector exists in 50 years. We must remain pragmatic in our thinking. The work conducted over the next two days will help us to move beyond second-generation nuclear technology, maintain economic competitiveness, and convince the public that nuclear energy is acceptable in the middle and very long term. We must also show imagination: the innovative solutions have certainly not proved themselves yet. What will a nuclear reactor be like in 50 years: the second generation of PWR; a rapid neutron reactor; a hybrid reactor? Many solutions may be envisaged, but much research remains to be done to confirm their validity. What will be the solutions for the back-end of the fuel cycle in 50 years time? In less than 10 years now, as required by the law of 30 December 1991, France will review the various options available: separation, incineration, very long-term storage and disposal – all matters that call for imagination and realistic innovation.

In conclusion, I would like to thank the OECD Nuclear Energy Agency for having chosen to hold this workshop in the Rhône valley. Nearly a thousand years ago Avignon was the home of popes. It was a place where ideas were discussed, a place of conflict, and an international centre. A thousand years later, however, it has become an historic city. I do not claim to believe that our work will have the same impact as earlier historical events, but will they be imbued with the same passion?

Tomorrow you will have the opportunity of visiting the Atalante laboratories. Atalante too is a name full of mystery. Perhaps we alone, however, have the responsibility of going beyond the myth. For the CEA, this facility symbolises our determination to transform our research programmes into realities. It is the instrument of our ambition to see the scenarios we study take concrete form. But above all, I would like you during this visit to meet our researchers, engineers, and technicians who, armed with their expertise and their faith, strive from day to day for the success of our common enterprise.

I wish you fruitful discussion, and realistic and encouraging conclusions. And perhaps you will benefit from the mild climate of the Rhône valley, which is so conducive to intellectual creativity.

SESSION #1

BACKGROUND, ISSUES AND CHALLENGES

CHAIRMAN: PROF. C.-O. WENE, SWEDEN

DRIVERS TO A 1 000 GWE NUCLEAR CAPACITY IN 2050

Evelyne Bertel
Nuclear Development Division, Nuclear Energy Agency

Hans-Holger Rogner
International Atomic Energy Agency

Introduction

Nuclear power development over the next five to six decades will be affected by a number of factors whose evolution is difficult to anticipate. Factors specific to nuclear technology, as well as other factors that relate to the overall energy and electricity markets at the international, regional and national levels, will impact on decisions to undertake or pursue nuclear power programmes in each country.

Nuclear scenarios to 2050 and beyond are developed by the Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA) in the context of a joint project on the potential role of nuclear power in sustainable energy strategies. The scenario chosen for the Workshop – 1 000 GWe in 2050 – corresponds to one of the alternatives considered within this project.

The main objective of nuclear scenarios is to serve as a backdrop for analysing alternative development paths and their consequences. The illustrative scenario adopted for this Workshop is intended to provide a framework for identifying issues arising with regard to uranium and fuel cycle service supply, including waste management and disposal, and to investigate how various back-end of the fuel cycle strategies could address those issues. This scenario, like any other, is by no means predictive but it is thought to represent a plausible future.

The paper presents briefly the status and trends of nuclear power programmes that served as a basis to develop the scenario to 2050. It elaborates on the factors that will affect nuclear power development and the conditions that could lead to a world-wide installed nuclear capacity of some 1 000 GWe by 2050. Global economic issues as well as factors specific to nuclear power technology and economics are discussed. Finally, some quantitative data on uranium and fuel cycle service requirements corresponding to this scenario are presented in order to identify key issues and challenges regarding the back-end of the fuel cycle.

Nuclear Power Today

At the beginning of 1998, there were nearly 440 nuclear reactors in operation in 32 countries, with a total capacity of around 350 GWe [1]. In 1997, nuclear power plants generated 2 276 TWh, which accounted for 17 per cent of the electricity produced world-wide and almost 6 per cent of total

commercial primary energy used. In OECD countries [2], where some 86 per cent of the world's nuclear power capacity is located, the nuclear share in total electricity generation was 24 per cent in average and exceeded 30 per cent in 8 countries.

Nuclear programmes have slowed down in the last decade or so as compared with the rapid development experienced in the mid-1980s. Only three reactors were connected to the grid in 1997, two in France and one in the Republic of Korea. Five reactors started construction during 1997, three in China and two in the Republic of Korea. Around 35 reactors were under construction at the beginning of 1998 in 14 countries, including 4 OECD countries, representing a total capacity of some 27 GWe. However, some of the units accounted for in this total might not be completed in the coming decade, if ever.

Several factors have led to a slow pace of nuclear power development. Lower than expected growth of electricity demand, especially in OECD countries, has reduced the need for additional power capacity and thereby orders for new nuclear units. Also the competitive margin of nuclear power versus fossil-fuelled plants has been reduced owing to drastic decreases of fossil fuel prices and technology progress, especially with regard to combined cycle gas turbines. Last but not least, socio-political barriers to nuclear power have increased in a number of countries.

Taking into account the lead-time for decision-making and implementation of nuclear power projects, and the retirement of old nuclear units, the nuclear capacity will increase slowly to 2000. The long-term scenario presented below assumes that the world nuclear capacity will be around 435 GWe in 2010 [3].

Factors Affecting the Future of Nuclear Power

The performance of nuclear reactors and fuel cycle facilities available on the market will be an important factor in the development of nuclear power. Key issues affecting the future of nuclear power include competitiveness, safety, radioactive waste management and disposal, and non-proliferation. Public acceptance, which is a prerequisite in democratic countries, depends largely on addressing those issues in a satisfactory way from the public's viewpoint.

However, future nuclear policies will be integrated in overall energy policies that are influenced by global economic and social factors. Whether or not nuclear power will continue to be developed will obviously depend on national choices and policies. Each country has a specific social and political context that shapes its energy policy and that may or may not be favourable to nuclear power development. Nevertheless, some key factors that are likely to influence nuclear policies are similar throughout the world.

Factors specific to nuclear power

Economics

Economic competitiveness remains the cornerstone for decision-making in the power sector. In the early 1980s, when most nuclear power programmes were launched, nuclear generated electricity was cheaper than fossil fuel generated electricity by a fairly large margin in many countries. The cost structure of nuclear electricity – capital costs represent around 50 to 70 per cent of the total while fuel cycle costs account for some 20 per cent [4] – was viewed as guaranteeing the stability of generation costs during the entire lifetime of a nuclear power plant and, thereby, ensuring stable electricity prices over a period of some four decades.

Today, if direct costs only are considered, a nuclear power plant is seldom the least cost option for incremental generating capacity. In most countries, low coal and gas prices prevailing on international markets, coupled with enhanced efficiency of coal and gas plants, allow fossil fuels to compete favourably with nuclear power for the plants to be built and connected to the grid in the coming decades. Gas-fired power plants, in particular, are an emerging competitor for baseload generation and, due to their low capital costs, fit the requirements of a newly deregulated and privatised market better than nuclear plants.

Nevertheless, there are some indications that nuclear power may regain competitiveness, even if only direct costs are to be considered. Cost reductions have occurred with regard to uranium and fuel cycle services. A number of nuclear units currently in operation have very low marginal electricity generation costs, and those costs tend to decrease as a result of additional feedback from experience and enhanced efficiency of operation. The experience acquired in operating plants, in France and the Republic of Korea for example, shows that nuclear power plant investment costs can be reduced by standardisation and efficient management of programme implementation. New reactor designs are aimed at reducing plant capital costs, which represent the largest share of nuclear electricity generation costs. Small and medium power reactors under development will provide more flexibility in nuclear investments and broaden the potential market of nuclear power. More generally, there is no reason to assume that nuclear technology, which is relatively young, cannot achieve additional performance improvements leading to further cost reduction while at the same time further enhance operating safety aspects.

Also, lifetime extension could improve the competitiveness of nuclear power plants. The experience acquired in replacing major components of operating units suggests that significant extensions are feasible. The technical lifetime of most operating reactors in OECD countries was initially expected to be 40 years. Today, many operators consider that these units will be operated for 50 years or more. This has a direct bearing on the cost of electricity produced. Finally, the new generation of nuclear power plants being developed or already under construction, such as the European Pressurised Water Reactor (EPR), is expected to have average availability factors above 85 per cent and thereby lower levelised lifetime generation costs.

The longer-term perspective for the competitiveness of nuclear power could improve dramatically if external costs are factored into decisions about new generating plants. One important external cost associated with sources of producing electricity relates to security of supply. This factor, which was considered extremely important 20 years ago, seems to have almost lost its importance for many countries. However, the analyses of the International Energy Agency (IEA) conclude that, in the absence of policy measures aiming specifically at alleviating dependence on OPEC oil, the share of OPEC in world oil supply will grow from the current 40 per cent to 50 per cent or more by 2010 [5]. As well, the sustainability of the “rush to gas” might be challenged on the grounds that the bulk of presently known gas reserves is located in regions with potentially fragile geo-political stability. This, and the fact that the expansion of gas supply infrastructures requires considerable lead times may – if the gas rush continues – well lead to temporary supply shortages and gas price volatility.

Another significant external cost is the potential economic impact of greenhouse gases. There are many widely varying estimates available on the costs of global climate change. What is clear is that greenhouse gas emissions from the nuclear chain are almost negligible. Full energy chain analysis shows that greenhouse gases from nuclear power and renewable energy systems are similar and that they are 40 to 100 times lower than fossil fuel chains for electricity generation [6]. Policy measures aiming at reducing carbon dioxide emissions through carbon taxes, emission limits or tradable permits would de facto enhance the competitiveness of nuclear power.

Safety

A general perception that there is a high level of safety in existing plants is vital to maintaining the viability of the nuclear power option. The Chernobyl accident made it abundantly clear that a nuclear accident anywhere affects nuclear power programmes everywhere, and was a major setback to further development of nuclear power in most countries. However, the safety record of nuclear power, particularly in the OECD, has generally been very satisfactory. In fact, an NEA *Collective Opinion* published two years ago reached just such a conclusion with respect to OECD facilities [7]. This record has been achieved through a defence-in-depth approach, extensive nuclear safety research programmes based on feedback from operating experience and enhanced by international co-operation through organisations like the NEA and the IAEA, and rigorous controls applied by independent regulatory bodies. Safety standards in non-OECD countries have not always measured up to standards in OECD countries. International co-operation and assistance programmes put into place since the Chernobyl accident, the broad adoption of safety culture world-wide, and the entry into force and implementation of the Nuclear Safety Convention, have improved the situation considerably. This high level of safety can be maintained and even enhanced by reactors under development.

Radioactive waste disposal

Perhaps the biggest challenge for nuclear power from the standpoint of public acceptance is to demonstrate that radioactive waste can be disposed of in a way to ensure that humans and the environment will not be harmed in the distant future.

Comprehensive comparative assessments that have been carried out on health and environmental impacts of nuclear power and other generation sources show that solid waste arising from the nuclear chain is small in volume compared to alternative sources. For example, in addition to 6 million tonnes of carbon dioxide, a 1 GWe coal-fired plant generates each year some 350 000 tonnes of solid waste, containing heavy metals and chemicals that remain toxic indefinitely, which is disposed of at surface sites from where it can migrate elsewhere. A nuclear plant of the same capacity and its supporting fuel cycle facilities generate about 500 tonnes of low-level waste, 200 tonnes of intermediate-level waste, and 25 tonnes of high-level waste when operated with a once-through cycle [8].

Considerable experience has been gained in the past few years in minimisation of operational waste, handling, treatment, storage and disposal of low and intermediate-level waste, conditioning (vitrification) of high-level waste, and storage of high-level waste and spent fuel. Short-lived low-level and intermediate-level radioactive wastes are disposed of in surface or near-surface repositories, already in operation in many countries and programmes are under way, although at a slow pace, for the implementation of deep geological repositories for long-lived high-level waste and spent fuel. Two *collective opinions* published by the NEA confirm that safety assessment methods are available to evaluate the potential long-term impact of radioactive waste disposal systems on humans and the environment [9] and conclude that the geological disposal strategy can be implemented in a manner that is sensitive to fundamental ethical and environmental considerations [10]. The bottom line is that, although experts agree that technical solutions exist for the safe handling and disposal of all types of radioactive waste, their implementation will only be possible if and when they will be accepted by the public.

Non-Proliferation

The potential non-peaceful use of nuclear materials and technologies is a concern, and the risk of proliferation of nuclear weapons is indeed one of the argument often used to simply write off the

nuclear power option. Proliferation risks have long been recognised and addressed by the international community and measures have been put in place to prevent, in so far as feasible, diversion of fissile materials. It is important to note that spent fuel from commercial nuclear power reactors contains only limited amounts of weapons-grade material not readily adaptable for weapons production even where the ability to separate it from spent fuel exists. The production of viable weapons from spent fuel would require large-scale, sophisticated efforts including chemical processing and handling procedures which, while within the potential reach of a handful of Governments, are virtually impossible for terrorists.

To eliminate the production or diversion of weapons-grade materials, the permanent Treaty on the non-proliferation of Nuclear Weapons (NPT) of 1970 commits 185 countries to refrain from acquiring nuclear weapons and to accept comprehensive IAEA safeguards on all their nuclear activities. A number of additional international agreements such as the Euratom and Tlateloco Treaties complement the international effort to monitor and physically protect nuclear materials associated with the peaceful use of nuclear power. The international safeguard regime, which has gained in strength in recent years, has proved to be effective in providing the assurance the world needs that proliferation remains in check.

The future of nuclear power depends on the current regime remaining strong and adapting, as necessary, to changing technological and political developments and evolving threats, including illicit trafficking in nuclear materials. One reason for the strength of the nuclear non-proliferation regime is that it has been buttressed by achievements in the field of disarmament, including the recent adoption of a comprehensive test ban treaty by the United Nations. In particular, the dismantling of large numbers of nuclear weapons in the United States and Russia, made possible in part due to the change in the geopolitical situation, is a sign that the threat of nuclear war has been greatly reduced. Also, in order to reduce even further the risks of proliferation, work has been started on the design of diversion-resistant reactors and fuel cycle systems.

Other factors

Globalisation

Globalisation of the world economy is resulting in a rapid growth in international trade and investment. National energy policies are increasingly influenced by the international situation and the policies adopted by other countries. In the energy sector, globalisation is perceived as enhancing security of supply and price stability in international markets. Free exchanges of energy and electricity across borders are reducing the incentive for energy policies aimed at national independence. Therefore, in so far as energy independence has been a factor underlying the implementation of nuclear power programmes in OECD Europe, globalisation is generally perceived as a factor that might reduce the incentive to develop nuclear power. At the same time, however, it should be recognised that the growth in international investments offers opportunities for financing the high capital cost of nuclear units through multinational funding.

Economic deregulation

Deregulation of the electricity market is progressively reducing monopolies and captive markets, thereby, putting pressure on producers to achieve enhanced competitiveness. In addition, deregulation implies greater uncertainties with respect to future market share for electricity producers which, in

turn, favours flexible expansion strategies based upon short-term demand projections, sales guaranteed by contracts, and small size power plants. At the same time, however, the opening of markets offers incentive to invest in large generation units, such as current nuclear power plants which can supply a broader range of consumers. The extensive French electricity exports to a number of countries, which are essentially made possible by nuclear generation, illustrate this possibility.

Privatisation of the power sector

Privatisation, which will transfer risks and costs from taxpayers to shareholders, is expected to lead to pressure for more efficient energy production systems. This factor will change the criteria of potential investors in the power sector. Large state-owned companies are expected to be replaced progressively by smaller independent producers for whom large, capital-intensive facilities will appear less attractive. Private investors and independent producers, having limited assets, are not likely to be attracted by large investments that may require 20 years or more to be amortised. As compared to alternative technologies, nuclear power plants require larger up-front investment and longer construction times during which the investor does not benefit from revenue. Nuclear power projects are essentially long-term, and uncertainty in electricity demand over a period of several decades introduces a financial risk that private investors might not be willing to take. Therefore, the privatisation of the electricity sector may tend to jeopardise potential nuclear power projects, even though they would be economically viable in terms of lifetime generation costs.

A 1 000 GWe nuclear scenario and its consequences

The nuclear scenario described below assumes a continued nuclear power growth in the context of energy strategies aiming at sustainable development. More specifically, it is consistent with the "ecologically driven" case described in the 1995 IIASA/WEC study on long term energy demand and supply [11].

This case is based on rather optimistic and challenging assumptions. It assumes that energy policies would integrate explicitly environmental protection objectives. A steady technological progress would enhance energy efficiency. The energy intensity of the world economy would decrease at an average rate of 1.4 per cent per year owing to policy measures and a broad deployment of innovative means to produce and use energy. Other assumptions adopted in this case include that world population will reach around 10 thousand million inhabitants in 2050 and that the economic growth will be moderate but accompanied by significant technology adaptation and transfer from industrialised to developing countries and reduction of present economic disparities.

In 2050, the world primary energy use would reach some 14 Gtoe involving some 23 000 TWh of electricity supply. Within this primary energy demand case, the continued growth nuclear scenario assumes that nuclear power programmes would continue in countries where nuclear units are already in operation and would be launched in countries which currently are planning to implement nuclear units by 2010-2015 [12]. Nuclear units reaching retirement would be replaced by new nuclear units. As a result, nuclear power capacity grows steadily but not at a very high rate because total energy and electricity demand growth rates are moderate and the nuclear power share in total energy supply increases slowly, reflecting economic competition from other electricity generating options and long lead times to implement nuclear power programmes. Nuclear electricity generation in the world would reach 7 850 TWh in 2050 as compared with 2 276 TWh in 1997. In 2050, nuclear would supply some 35 per cent of total electricity consumption corresponding to about 12 per cent of total primary energy demand, as compared with some 17 per cent and 6 per cent respectively in 1997.

Natural uranium requirements would depend on the reactor and fuel cycle strategies adopted. Assuming that reactors would be fuelled by uranium and operated on the once-through cycle, and that enrichment plant tails assay would remain at the present level of 0.3 per cent, annual uranium requirements would grow from less than 60 000 tU/y around the year 2000 to 175 000 tU/y in 2050. Those requirements exceed both the present level of production of fresh uranium (slightly more than 36 000 tU/y in 1996) and the production capability expected to exist early in the next century (below 60 000 tU/y) [13].

However, demand growth would be likely to stimulate an expansion of production capacity, as was the case in the late 1970s. Also, at present uranium supply is met partly by drawing from excess civil inventories, and this is expected to continue in the coming five to ten years. Moreover, in the medium term, dismantling of nuclear weapons will provide additional supply of fissile materials for power reactors.

On the demand side, uranium consumption per kWh can be reduced by: increasing fuel burn-up (thereby producing more energy per unit of nuclear fuel); lowering enrichment plant tails assays (thereby recovering more of the ^{235}U present in natural uranium); and recycling plutonium and uranium recovered from reprocessed spent fuel (thereby reducing the needs for fresh natural uranium) [14]. Also, in the long term, thorium could become an additional source of nuclear fuel if and when alternative reactor technologies would become commercially available.

Cumulative uranium requirements would reach 5.6 million tonnes of uranium in 2050 if all reactors were operated on the once-through fuel cycle and enrichment plants would operate at 0.3 per cent tails assay throughout the period. With those assumptions, present uranium reserves (reasonably assured resources recoverable at less than US$ 80/kgU) would be exhausted by 2025 and presently known uranium resources would run out by shortly after 2040. However, the cumulative uranium requirements would be far below total conventional resources recoverable at less than US$ 130/kgU (around 16 million tonnes U) [13]. Within a period of several decades, with additional exploration efforts, a significant part of the known uranium resources could become reserves and additional resources could be discovered. In response to growing demand and rising uranium prices, exploration efforts and new mine developments would be possible.

Also, as mentioned above, uranium requirements could be reduced significantly by reducing enrichment plant tails assay and/or reprocessing spent fuel and recycling the recovered plutonium and uranium. Lowering enrichment plant tails assay from 0.3 to 0.15 per cent would reduce cumulative uranium requirements by 2050 from 5.6 to 4.2 million tonnes U. Reprocessing all light water reactor (LWR) spent fuel and recycling the uranium and plutonium in mixed-oxide fuel (MOX) for light water reactors (loaded with 30 per cent MOX and 70 per cent uranium oxide fuel) would lead to a cumulative saving of some 600 000 tonnes of natural uranium by 2050. The combined effect of lowering tails assay and recycling would reduce cumulative uranium requirements by more than 30 per cent.

Spent fuel arisings would increase steadily if all reactors would be operated on the once-through fuel cycle, reaching nearly 19 500 tHM/year by 2050, i.e. more than twice the 1995 annual spent fuel arisings (around 9 300 tHM). Reprocessing and recycling strategies would reduce significantly non-reprocessed spent fuel arisings. Assuming that all LWR spent fuel would be reprocessed and recycled in LWRs accepting up to 30 per cent MOX in core, non-reprocessed spent fuel arisings in 2050 would be reduced to around 5 000 tHM/y, i.e. less than half of the arisings in 1995.

In that strategy, reprocessing requirements would reach around 8 000 tHM/y in 2025 and 15 000 tHM/y in 2050, and MOX fuel fabrication requirements would be around 1 000 tHM/y in 2025

and 1 900 tHM/y in 2050. The existing and planned capacities for reprocessing LWR fuel and for fabricating MOX fuel assemblies could meet the requirements during the first two decades of the next century, but new capacity would be needed by 2020. The introduction of fast reactors could reduce even further, and eventually eliminate, the accumulation of non-reprocessed spent fuel and of plutonium in excess of hold-up inventories at reactors and fuel cycle facilities.

Concluding remarks

An analysis of the factors affecting nuclear power development shows that a 1 000 GWe nuclear scenario by 2050 is plausible. Although it does correspond to present trend extrapolation, it may result from energy policies aiming to sustainable development and would help in addressing global climate change issues as well as other environmental concerns.

Whatever the development of nuclear power will be, optimising back-end of the fuel cycle strategies will be essential to maintain the nuclear option viable. Indeed, even in a phase-out scenario, it would be necessary to ensure safely the decommissioning of nuclear facilities and the disposal of radioactive waste. In a continued moderate growth of nuclear power scenario, back-end of the fuel cycle strategies would have to address a broad range of issues related to natural resource management and environmental protection as well as economic optimisation. For example, technologies should be implemented at a large industrial scale for spent fuel storage and disposal and/or plutonium handling, transport and utilisation.

The papers that will be presented and discussed during this Workshop demonstrate that extensive R&D activities are on-going world-wide to investigate alternative back-end of the fuel cycle strategies that could meet the objectives of sustainable development. International organisations such as the NEA and the IAEA offer a forum for exchange of information and experience sharing and may assist Member countries through this process by creating a synergy enhancing the efficiency of national efforts.

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ISSUES AND CHALLENGES

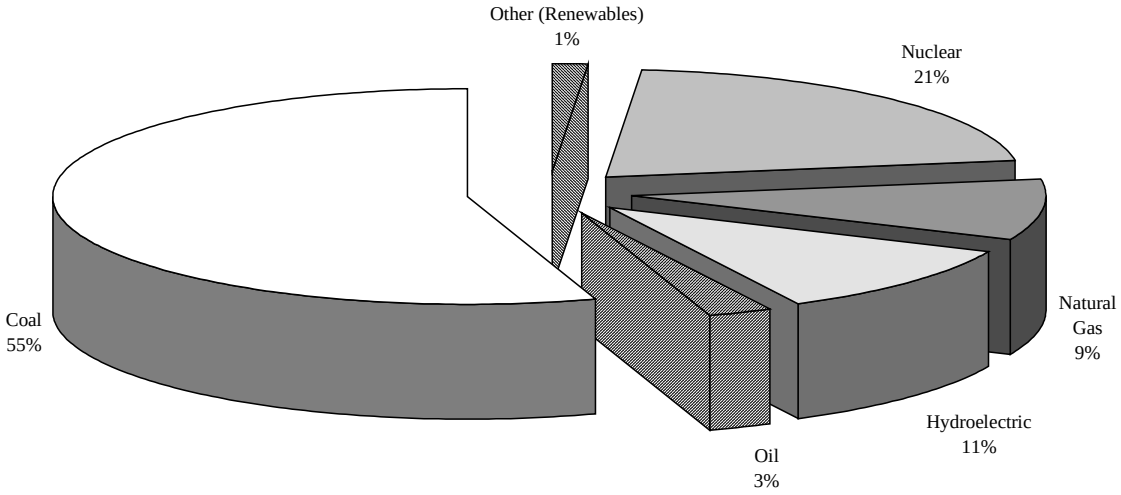
Trevor Cook
US Department of Energy, USA

NUCLEAR ENERGY IN THE UNITED STATES

Nuclear Energy is a Vital Component of US Energy Mix

Today, 105 commercial nuclear power plants produce more than one-fifth of US electricity.

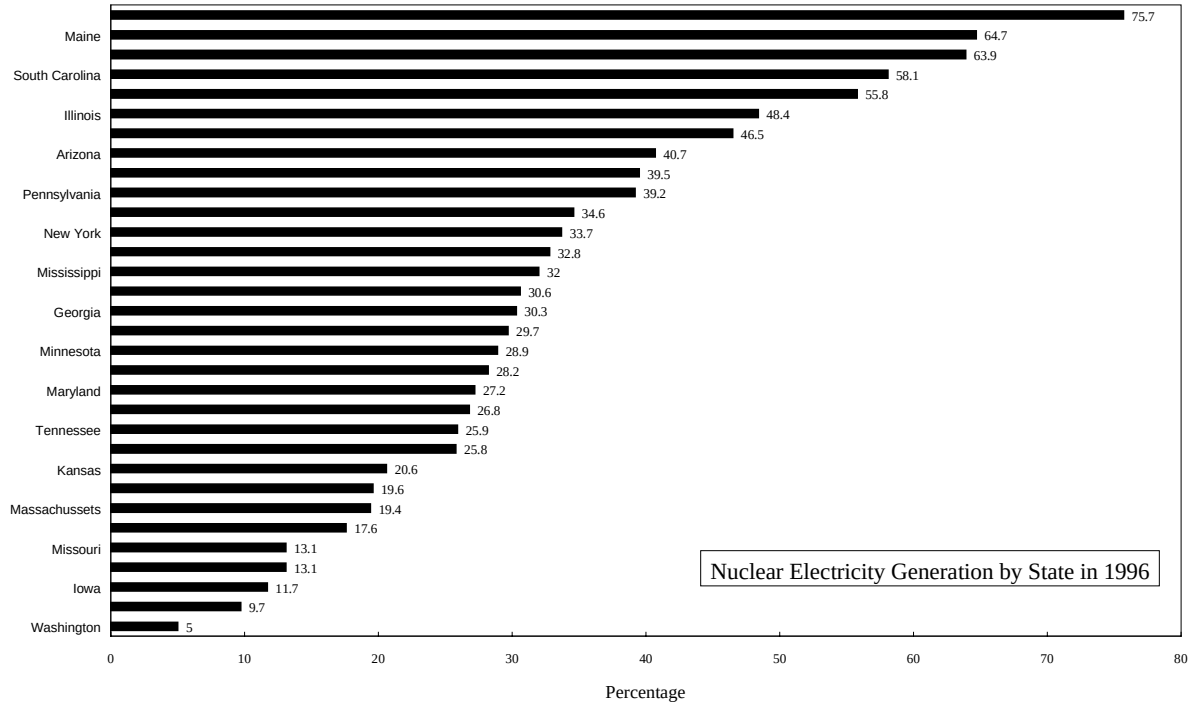
Fuel Share of US Electric Generation in 1996



Source: EIA Annual energy Outlook 1998

- Nuclear energy provides reliable baseload electricity in all weather conditions
- Most US nuclear power plants have low production costs and can be competitive sources of electricity

Many US states rely on nuclear power for a large portion of their electricity requirements

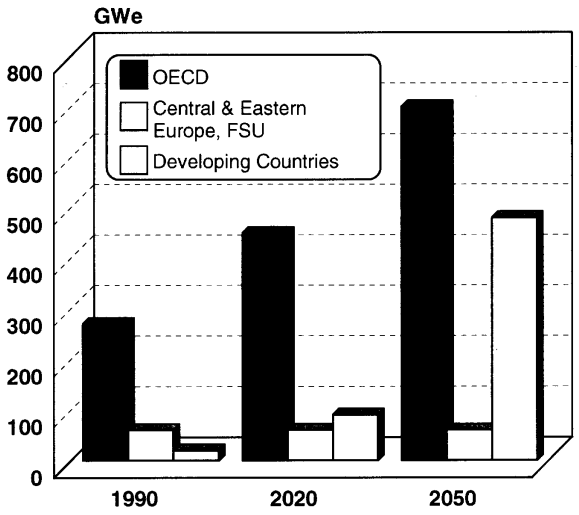


Source: EIA 1997

- US nuclear plants generate over 100 gigawastts of electricity annually
- Large baseload electricity source second only coal

WHAT ROLE WILL NUCLEAR ENERGY PLAY IN THE FUTURE?

Use of Nuclear Power Is Expanding in Many Countries



Reasons For Expansion

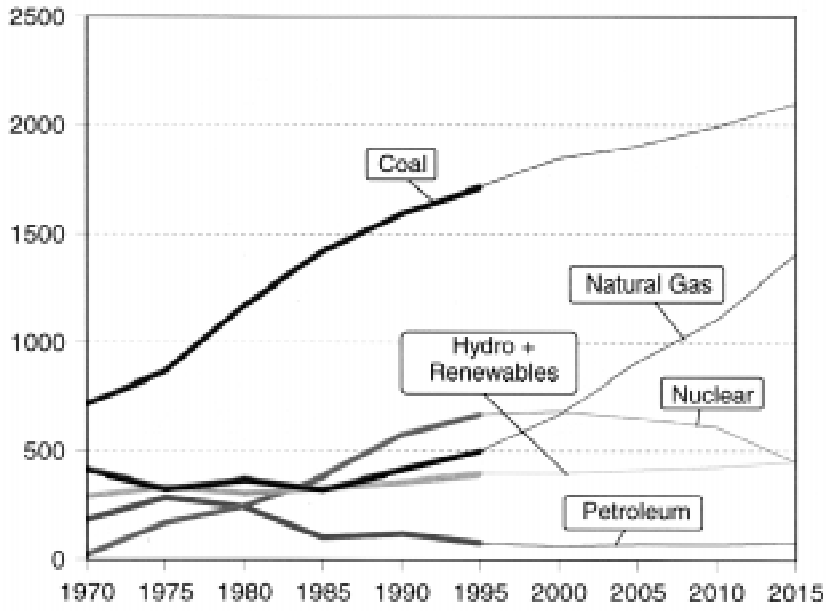
- Energy security
- Lack of extensive fuel resources and transportation systems to support coal or natural gas plants
- Concerns over air pollution associated with economic growth

1. ECONOMIC COMPETIVENESS

1.1 Issues and Challenges

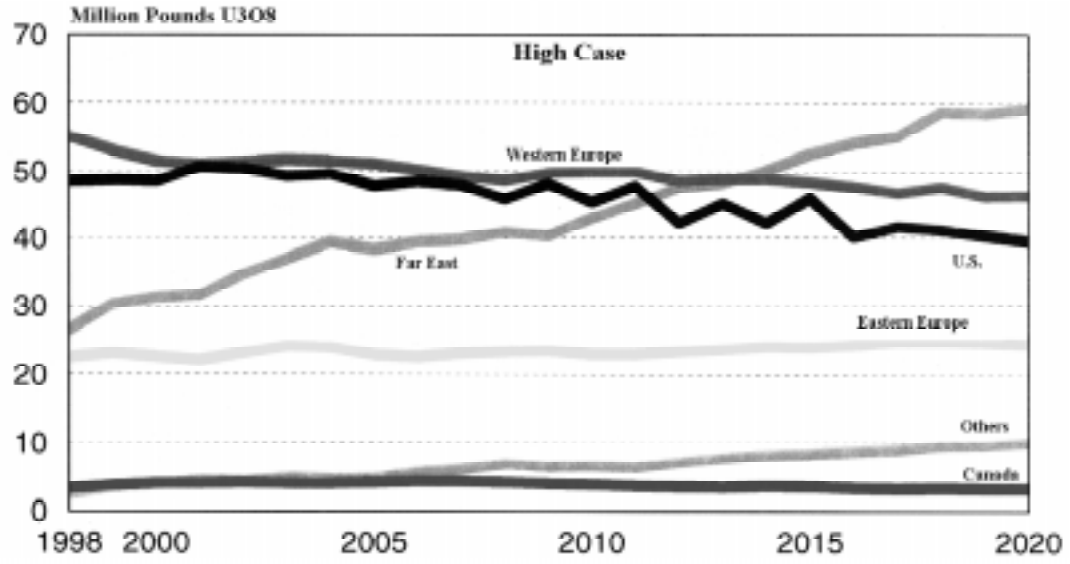
- Economic Competiveness
- Environmental Quality
- Weapons Proliferation
- Decline of the Infrastructure

1.2 Projected US Electricity Generation by Fuel



Source: EIA Annual Energy Outlook 1997

1.3 Projected World Annual Uranium Requirements



Source: Nuclear Power Generation and Fuel Cycle Report 1998 published by US DOE Energy Information Agency

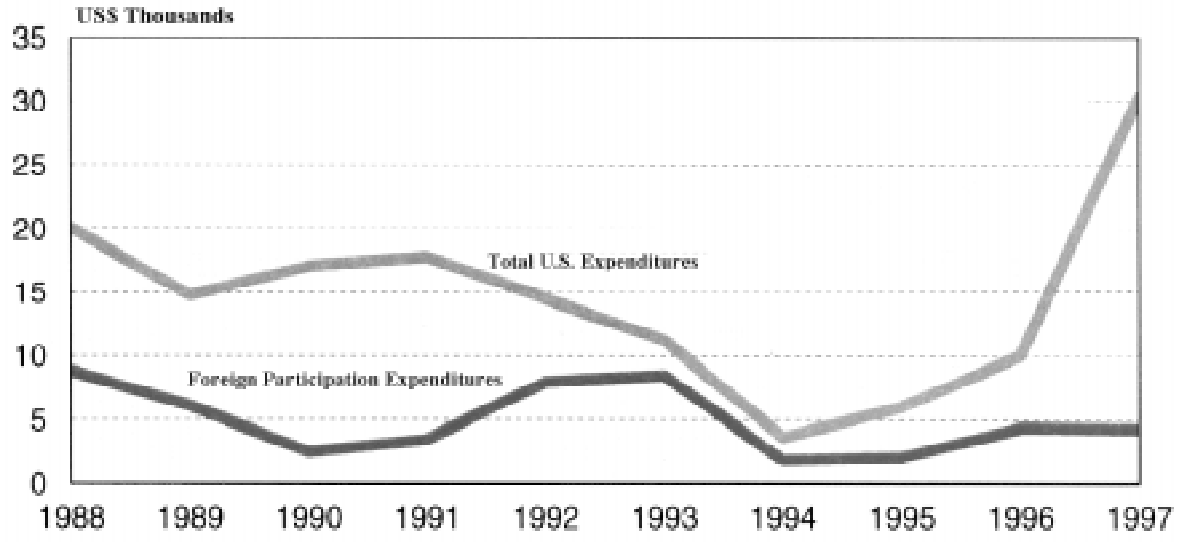
1.4 Forward-Cost Uranium Reserves By Mining Method, 1997

Mining Method	Forward-Cost Category U_3O_8 (million pounds)	
	$30 per pound	$50 per pound
Underground	139	465
Openpit	29	257
In-Situ Leaching	113	194
Other	<1	15
TOTAL	**281**	**931**

1.5 Inventories of Natural and Enriched Uranium

Type of Uranium Iventory	1997 Inventories
US Utility Inventories	63,936
Natural Uranium	45,874
Enriched Uranium	18,061
US Supplier Inventories	11,908
Natural Uranium	10,257
Enriched Uranium	1,652
Total Commercial Inventories	**75,844**
DOE-Owned and USEC-Held Inventories	102,929
Natural Uranium	76,542
Enriched Uranium	26,388

1.6 Expenditures for Exploration and Development of Uranium in the United States 1998-1997



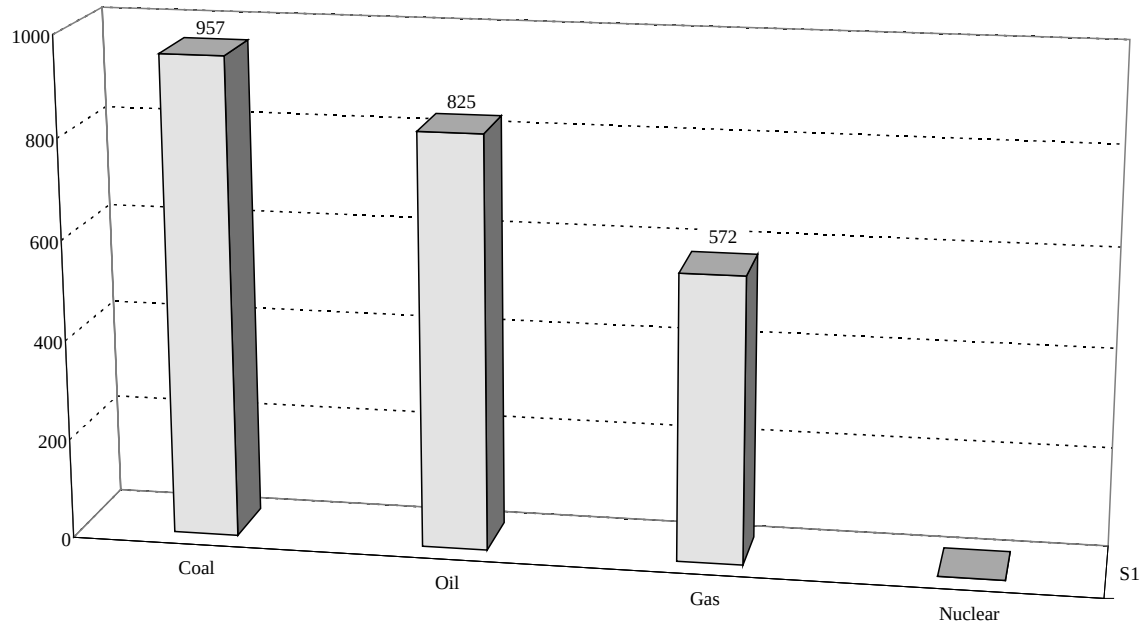
Note: Totals may not equal sum of components because of independent rounding.

Source: Energy Information Administration: 1998-1996-Uranium Industry Annual 1996 (April 1997). 1997-Form EIA-858, "Uranium Industry Annual Survey"(1997).

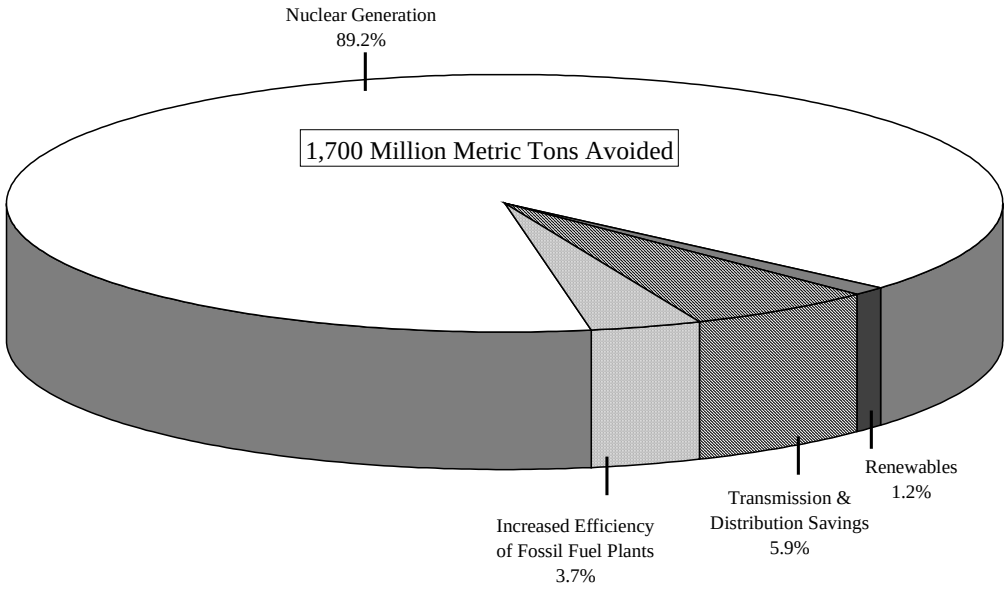
2. ENVIRONMENTAL QUALITY

2.1 Nuclear Energy Mitigates Global Climate Change

Nuclear power produces essentially zero carbon SO_2 or NO_X gas emissions.



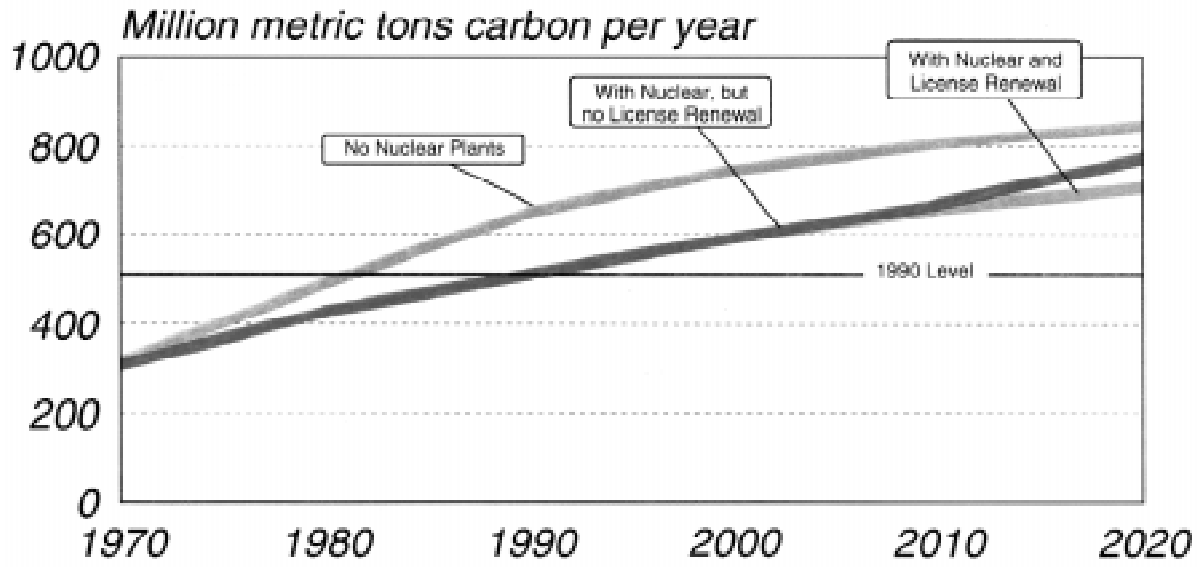
Nuclear power contribution to US carbon emission reductions*
(Carbon emissions avoided 1973-1994)



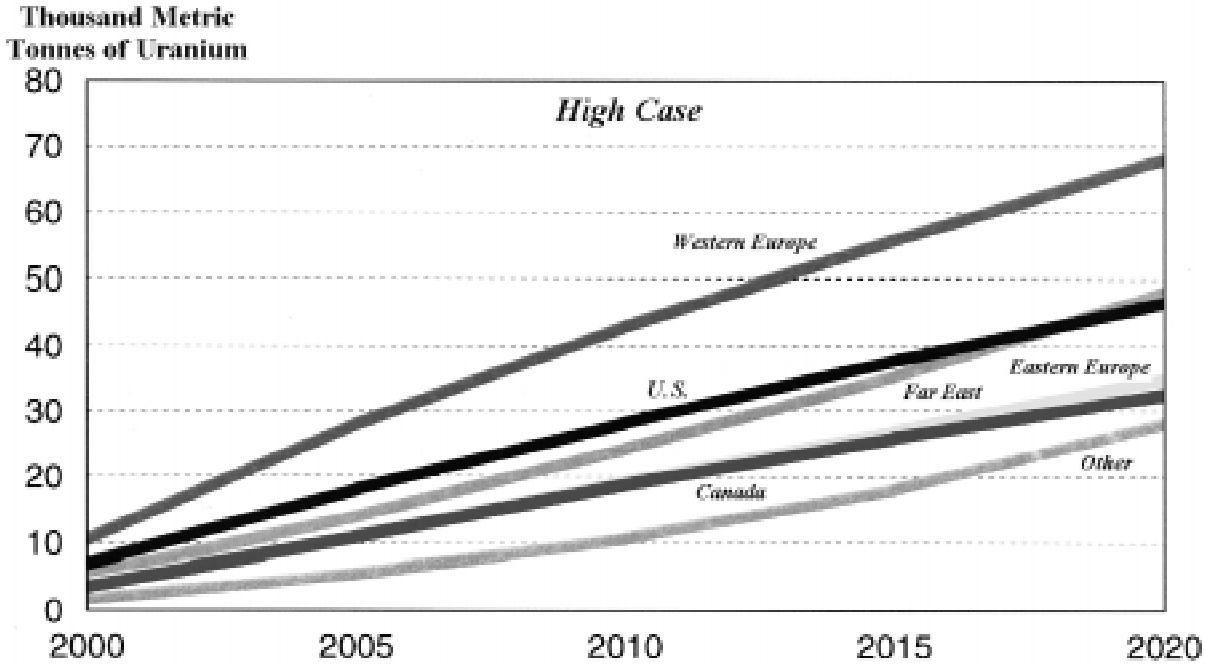
* Displacements are in million metric tons of carbon (C) weight

If 75 percent of US nuclear plants renew their license an additional 2.8 billion metric tons of carbon emissions will be avoided by 2035.

2.2 Impact of License Renewal on Electric Generation Carbon Emissions in the United States (1997 to 2020)



2.3 Projected World Cumulative Spent Fuel Discharges



Source: Nuclear Power Generation and Fuel Cycle Report 1998 published by US DOE Energy Information Agency

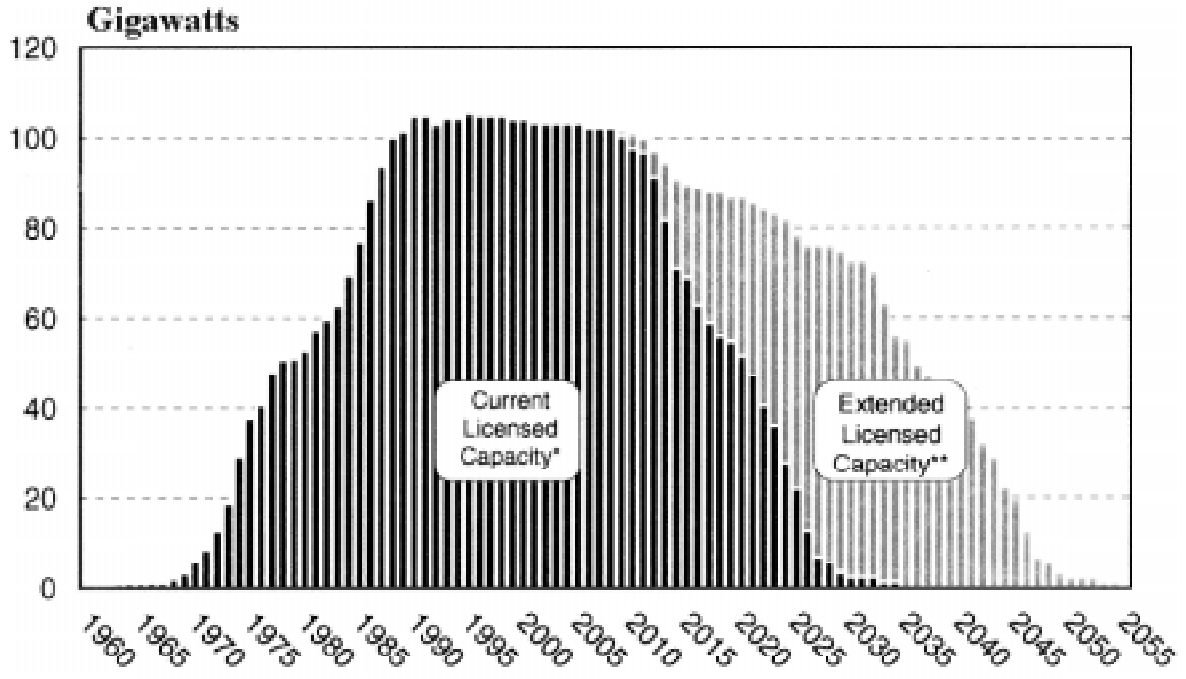
3. WEAPONS PROLIFERATION

- Proliferation resistant fuel
- Once-trough versus recycling technology
- Fast Breeder Reactors
- Thorium-Uranium Fuel Cycle

4. DECLINE OF THE INFRASTRUCTURE

4.1 Nuclear Plant License Renewal

- Success of current plants and prospects for future US plants depend upon accomplishing license renewal for a significant portion of US fleet



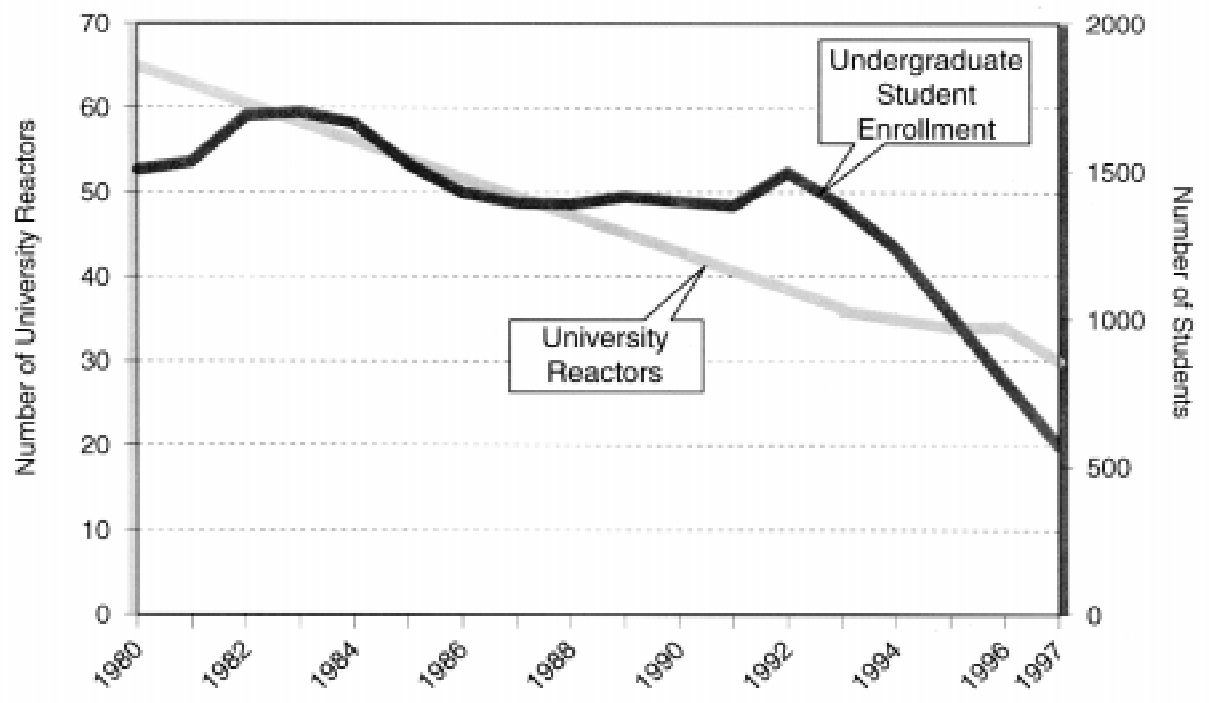
Source: DOE Analysis

* Assumes 5 percent of current plants are shut down before the end of their initial license period.

** Assumes 75 percent of plants receive license renewal for 20 years.

- Plan includes research to address generic issues, dissemination of information through the industry, and active participation of DOE in resolving issues with NRC

4.2 Negative Trends in University Nuclear Engineering



SESSION #2

EVOLUTION OF CURRENT TECHNOLOGIES

CHAIRMAN: DR. K. FOSKOLOS, SWITZERLAND

FUEL UTILISATION IMPROVEMENTS IN CURRENT REACTORS

by Jean-Luc Provost
EDF, Fuel Operations and Strategy Group

Fuel Management Strategy

Context

Since the commissioning of first industrial size PWRs in the seventies, core management has substantially progressed thanks to various advances in fuel technology. This progress has been sustained by operational feedback as well as R&D programmes on fuel design and methods perfecting.

The programmes have helped improve understanding of fuel behaviour and the conditions for its utilisation in reactors and, generally speaking, have resulted in fuller use of existing potentialities and the maintain of proper technical and safety margins.

The developments in fuels must be seen in the context of the nuclear power industry:

- the fuel product is at the interface of the nuclear material and the Nuclear Steam Supply System (NSSS). Improvements in the fuel therefore have to be measured in relation to NSSS adaptation capabilities;
- the fuel product and its utilisation in the reactor are merely stages in the nuclear fuel cycle. Any improvements, therefore, are an integral part of the fuel cycle strategy in general, and back-end fuel cycle management, in particular;
- the fuels should benefit from significant advantages from the standardisation of the utilities' fleet of reactors (EDF for example, with 57 PWRs using very similar fuel assemblies);
- fuel costs (front-end and back-end) account for around a quarter of the cost of producing a unit of nuclear-generated electricity.

Strategy

The strategy for the development of fuel products and their utilisation in reactors is based on a careful and gradual use of identified margins and technological breakthroughs on the fuel and on the NSSS to achieve, within the framework of safety requirements, the following objectives:

- improve the fuel reliability in the reactor;
- reduce radiation exposure of staff;
- reduce electricity production unit cost.

The key objective to reduce electricity generating cost involves minimising the sum of the following two costs:

- fuel cycle cost: to draw maximum benefit from improvements in fuel performance, in particular discharge burnup increase, and to utilise the fissile materials arising from reprocessing by recycling Enriched Reprocessed Uranium (ERU) and plutonium (MOX); and
- overall generating system management cost: depending on numerous parameters as electricity demand level over the year (seasonal outage scheduling), replacement energy cost (hydro, fossil), plants maintenance, plants operational flexibility. This cost decreases when cycle length increases mainly due to higher plant availability.

Extended length cycles

In pursuing this strategy, utilities have generally given top priority to the extended length cycles, an objective dictated by the search for:

- higher safety level based on a reduction in the number of annual outages on each site. This in turn improves preparation and monitoring of operations;
- lower overall radiation exposure (radiation levels being highest during outages);
- higher availability, hence lower operating costs.

For example, from 1996, EDF began the introduction of 3 batch-core management with 4 per cent enrichment in all 1 300 MWe PWR reactors (necessitating the use of gadolinia fuel rods to control reactivity at the beginning of the cycle). This management will make it possible to extend cycles to 18 months. Similar changes are planned for the six oldest French 900 MW PWRs (FESSENHEIM and BUGEY). It is expected that, by the early 2000s these reactors will be running on 3 batch-core management using uranium enriched to 4.2 per cent. This will increase the cycle from 12 to 16 months.

Recycling of fissile materials produced by reprocessing

Some of the main nuclear power generation utilities in Europe have chosen to close the fuel cycle. In France, the reprocessing/recycling strategy, based on the equality of flows, has led EDF to limit reprocessed volumes in such a way that only immediately recyclable quantities of plutonium are separated (adequation between reprocessing, manufacturing and recycling quantities).

Today in France, the MOX fuel manufacturing plant (MELOX) has attained its nominal capacity, securing the supply of the current EDF needs (100 tHM of MOX per year). From July 1998, 20 PWR 900 MWe are authorised to load MOX fuel and 15 of them have been loaded with this fuel. Another 8 will require their commissioning decree to be amended and to be subject to a public inquiry.

Regarding enriched reprocessed uranium (ERU), the recycling is undertaken on two PWR 900 MW operating with 3.7 per cent U235 enrichment (3.4 per cent ENU equivalent), in 4 batch-core management.

As regards the recycling of both plutonium and reprocessed uranium, EDF's aim is to attain parity between recycled fuels i.e., equivalence between the different core managements (enrichment, authorised burnup rates and cycle length identical to those for UO_2), absence of restrictions in terms of manoeuvrability, at least in equilibrium cycles, and management of transport and handling logistics at power stations. Such a parity reinforce on the economic sight the reprocessing/recycling policy.

The Current Situation

Fuel management

Today in France, the 34 PWR 900 MW operate on 12-month cycles, with 3.7 per cent enriched uranium in 4-batch core management and with MOX in hybrid management (3 batches for MOX and 4 batches for uranium). The 20 PWR 1 300 MW operate on 18-month cycles with 4 per cent enriched UO_2 in 3-batch core management.

The UO_2 average discharge burn-up is about 44 GWd/t, with certain assemblies reaching the authorised limit of 47 GWd/t or even exceeding it by virtue of a special permit (some assemblies reaching 50 GWd/t).

The MOX average discharge burnup is about 37 GWd/t, some assemblies reaching 40 GMWd/t.

Fuel performance

The fuel products in use today are of advanced fuels of second generation. These different products, designed in the early 90s, have produced four major benefits:

- reduction in the radiation doses received by the workforce thanks to the adoption of zircaloy grids;
- better fuel rod performance, higher cladding resistance, especially to corrosion, together with higher average burnup and maintenance of margins in relation to technological limits;
- general adoption of grids with higher thermohydraulic performance permitting the observance of margins in relation to the critical heat flux phenomenon;
- systematic use of anti-debris device permitting enhanced assembly performance.

These benefits have helped to maintain or improve the margins identified in relation to the fuel's technological limits.

Fuel behaviour

In France, since the commissioning of the first PWR in 1977 until end 1997 some 38 000 fuel assemblies were loaded into reactors. EDF's experience to date is based on 600 reactor cycles.

The adoption of MOX in the 900 PWR reactors is underway (16 reactors are opened today), and MOX operational experience is 70 reactor-cycles with 800 MOX assemblies loaded into reactors.

Reliability

The fuel's behaviour in 1997 was on the whole satisfactory in France, with a marked improvement in the cladding leakage ratio compared with 1995 and 1996. As a result, reactor activity remained well within the technical specifications limits. The fuel's failure rate was 0.14 per cent in 1997 against 0.26 per cent in 1996 and 0.41 per cent in 1995 (in per cent of examined assemblies at each outage).

The main cause of cladding leakage is the presence of loose parts in the primary circuit. The gradual adoption of anti-debris devices at the base of assemblies should help to attenuate this cause. A large part of the presently loaded assemblies are equipped with this device and in 1999 all assemblies will be fitted.

Handling incidents

Damage to assemblies may also occur when the fuel is handled during unloading/reloading operations. However, the number of assemblies damaged during handling has sharply dropped (20 assemblies in 1997). The main reason for this drop is the technological improvements to second-generation assemblies such as reinforced grid corners that are more shock-resistant. Furthermore, significant investments have been made by the operator in training, equipment and handling procedures.

Assembly deformation

In late 1995, as well as in 1996 and in 1997, the 1 300 MW reactors experienced a new type of incident: an increase in RCCA drop time. The main cause of these incidents is lateral assembly deformation which leads to greater friction between the control rods and the guide-tubes and then, in some cases, to incomplete rod insertion in the dashpot.

In order to overcome the problem, a thickness increase of the guides tubes has been implemented on all the EDF fuels which enables the structure rigidity to be increased. Other improvements to the assembly are under study like AFA-3G design from FRAGEMA.

Similar incidents occurred on PWR reactors in the USA, and Europe.

Reactor operation

Reactor manoeuvrability

Given the significance of PWR reactors in EDF's electricity output (80 per cent of electricity is nuclear generated) the reactors must contribute to the balance between production and consumption (frequency regulation, load follow, extended low power operation).

Moreover, to optimise the schedule of reactor shutdowns, greater reactor cycle flexibility is required. This will make it possible by anticipated shutdowns or stretch-out operations, which in turn would influence the burn-up rates of unloaded fuel.

These operational features, which are more fuel-constraining than base load operations (particularly in regard to the thermomechanical behaviour of the cladding and pellet-clad interaction), are taken into account in fuel design. All these types of operations, together with outages schedule flexibility, are currently authorised for the management adopted at all PWRs.

The chemistry of the primary circuit

The adoption of longer 18-month cycles for the 1 300 MW has been accompanied by the presence of markedly higher concentrations of boron in the primary circuit at the beginning of the cycle than those observed in the annual cycle. The boron may attain concentrations of about 1 500 ppm at the beginning of an equilibrium cycle.

The maximum concentration of lithium hydroxide (used to control the pH) is presently fixed at 2.2 ppm. This limit was fixed by the fuel manufacturer, by mutual agreement with EDF, for the purposes of protecting the rod cladding from corrosion. Instead, it results in a deviation from the optimum chemical characteristics of the primary circuit (constant pH of 7.2 at 300°C) because of greater release of corrosion products and increased radiation doses.

The imposition of a 2.2 ppm lithium limit in extended cycle operations causes the reactors to operate in non-optimum primary chemical conditions for several months, increasing the annual radiation doses for 15 per cent.

Tests are therefore being conducted this year at a reactor to raise the lithium limit to 3.5 ppm to reach in a shorter time the optimum pH level of 7.2 and thereby to reduce radiation doses to the initial values. A fuel monitoring programme is attached to this experiment in order to study changes in the fuel cladding in situations of high concentration of lithium. The levels in oxidation and hydridation of the cladding will be quantified.

Vessel fluence

The founding study set vessel life at 40 years on the basis of initial core managements: 3.25 per cent in 3-batch core management for 900 PWRs and 3.1 per cent in 3-batch core 1 300 PWRs using natural enriched uranium fuel. The loading patterns taken into account were of the out-in type and related to base load operation.

Today, both operating conditions and management methods have changed, particularly as a result of plutonium recycling and a switch to longer cycles.

In order to preserve vessel life, the optimisation of the core loading patterns involve a reduction in the fluence at the vessel hot spot. New assemblies shuffle rules have been set to minimise the reactivity of the assemblies exposed to the vessel hot spot (3-cycle or 4-cycle assemblies), with the loading pattern remaining out-in.

These rules, already in place in all PWR, have led to a significant reduction in overall vessel fluence on a 40-year life basis (the reduction can reach 20 per cent compared with origin estimates).

These measures have nevertheless caused a slight increase in radial peaking factors, disruption in the distribution of radial power and greater dispersion of assembly discharge burnup that could slightly increase the irradiation of the most depleted assemblies.

Observance of safety requirements

Although fuel safety criteria constitute the basis of the safety policy, they downsize the reactor power capability.

To ensure fuel rod integrity in the event of class 2 accidents, the following criteria apply:

- prevention of critical heat flux, reflected in compliance with a DNBR criterion. The fuel thus has to have a high performance in the face of the critical boiling phenomenon. This is the result sought via optimisation of mixing grid geometry, with performance being quantified by means of experimental loop tests;

- non-pellet melting, reflected in the compliance with a linear heat power rate of 590 W/cm. This criterion is not restrictive today;
- prevention of cladding failure by pellet/clad interaction, reflected by compliance with a much lower linear heat power rate than that applicable in non-pellet melting. The related technical specifications are restrictive given the special conditions of manoeuvrability applicable to PWRs place very severe thermomechanical loads on the fuel rods. As this criterion restricts reactor power capability, fuel suppliers are developing new products that will help to raise this limit.

The safety criteria applicable to the study of class 4 accidents have been re-examined by the Safety Authorities to take account of higher burn-up.

- To demonstrate fuel non-dispersion for high burn-up rods during a rod ejection accident, it was necessary to carry out several RIA tests (massive reactivity insertion) in the CABRI loop. Findings from the tests already carried out justified the fuel's compliance with the non-dispersion criterion in the primary circuit for the UO_2 and MOX fuels used in the current core managements.
- As regards LOCAs, the resistance of the cladding of highly irradiated rods during this transient needs to be demonstrated. Irradiated and oxidised cladding quench tests that have been conducted do not call into question the end-of-transient maximum acceptable oxidation limit of 17 per cent.

In conclusion, for 3.7 per cent 4-batch core management cycles at 900 MW PWRs, the current authorised assembly burnup limit is 47 GWd/t in France. Exemptions are, however, obtained every year for several assemblies exceeding 47 GWd/t provided they remained within an upper limit of 50 GWd/t.

The grant of a generic authorisation raising the assembly burnup limit to 52 GWd/t for 900 MW as well as 1 300 MW reactors is subject to investigation of issues raised by the Safety Authority especially RIA checks and LOCAs. EDF's aim is to obtain this authorisation in 1998.

Fuel Development Strategies

Guidelines

Principal fuel development objectives are as follows:

- maintain high fuel reliability while reducing the incidence of assembly deformation in the short term;
- obtain a product for existing reactors with better performance against pellet-clad interaction, and fewer operational constraints;
- raise the performance of fuels arising from reprocessing-recycling (MOX and ERU) so that they have the same energy and burnup as enriched natural uranium fuel;
- develop within the next ten years a fuel capable of achieving a burnup of 60 GWd/t, with a view to obtaining better economic optimisation of core management on EPRs (using uranium enriched to 5 per cent), without hampering reactor operation or compromising safety or fuel reliability. Some of the repercussions from this high burnup programme could be applied to existing reactors (24 months – 3-batch core management, annual – 5-batch core management).

Maintaining reliability

With regard to fuel reliability, the aim is to maintain the positive results achieved in 1997. Accordingly, the following measures have been adopted:

- Systematic quality control of all stages of the fuel manufacturing process at suppliers' and sub-contractors' plants. Implementation of regular quality audits to ensure compliance with procedures.
- Completion of the installation of anti-debris devices by 1999 (the first devices were introduced in 1992). This measure should all but eliminate rod failures at the first grid level, which has been the main cause of leakage in the last few years.
- Endurance tests to check the resistance of new assemblies to rod wear in the grids. It is difficult to represent this phenomenon in simulation models. Fretting was in fact responsible for several leakage incidents in the early nineties for some of the fuel suppliers.
- Reduction of the oxide layer thickness observed on the cladding (corrosion). The purpose of the new cladding materials is to reduce oxide thickness and to eliminate spalling. The materials having the greatest potential are alloys of zirconium containing niobium or vanadium. Different types of experimental cladding developed by fuel suppliers are currently being tested in PWRs.
- Strengthening of the assembly structure to solve the problem of incomplete rod drop observed recently on the PWR 1 300 MWe. The suppliers are studying various design improvements, including thicker guide-tubes, dashpot reinforcement, grid modifications, lighter hold-down springs, use of intermediate flow mixers, fuel rods on bottom nozzle, optimisation of structure materials, etc.

Relaxing PCI limits

Improving performance against PCI, whose limits place severe restrictions on PWR operations, involves the implementation of power ramps.

Fuel manufacturers are developing new products that should allow this limit to be raised. The focus of research is on cladding materials (slower creep), cladding geometry and pellet design (advanced microstructures, doping, geometry).

UO_2 and MOX parity

EDF's objective is to use MOX fuel equivalent to the uranium in four-batch core management by the early 2000s. MOX fuel will be equivalent to uranium fuel enriched to 3.70 per cent and could reach 50 GWd/t in the future. Studies on a such design are implemented up to now.

A number of experiments and studies are under way with a view to achieving higher discharge burnup rates and more economical core management:

- examination of irradiated materials to improve understanding of fuel rod behaviour;
- neutron model validation tests for high plutonium contents;
- fuel management studies (annual four-batch core management);
- optimising fuel rod design to accommodate greater fission gas release;
- NSSS safety studies assessing the impact of loading higher quantities of plutonium into the core on the efficiency of control systems during accidental transients.

Improving fuel performance

Improved fuel performance is primarily dependent on a significant increase in burnup beyond 52 GWd/t. Important design improvements also need to be made to both rods and structure.

As regards the behaviour of fuel rods in high burnup, the main aspects to be taken into account are:

- internal pressure due to the fission gas release (particularly for MOX);
- cladding corrosion;
- rod growth;

With regard to the behaviour of fuel assembly structure for high burnup, the following points should be examined:

- mechanical resistance of the grids and guide tubes;
- skeleton deformation;
- wear due to rod-grid friction;
- hydriding and corrosion.

The cladding resistance at high burnup will be greatly improved by the use of zirconium alloys, optimised in components and in the manufacturing process. However, the behaviour at high burnup of the optimised alloys supplied by the various manufacturers has still to be validated in PWRs.

The burnup resistance of the structure (wear caused by rod-grid fretting, deformation, etc.) has also been taken into account by the various manufacturers and should not in principle pose any problems. Some manufacturers have already brought out improved designs of guide tubes and grids (materials and geometry) in that goal. Nevertheless, the problem of assembly deformation recently observed in PWRs prompts us to remain cautious in this regard.

When the potential performance of new fuels is quantified, by means of irradiated fuel examinations and following the introduction of power ramps, these new fuels should help to make the technical specifications of current management more flexible.

Nuclear steam supply system adaptations

The adoption of new core managements for PWRs currently in operation can lead the operator to implement some adaptations of the nuclear steam supply system (NSSS), and more generally some modifications of the plant. For utilities, the decision to modify the plant is first an economical decision, in other words the only ones to be acceptable are light modifications on an economical and technical point of view.

The current studies of improved core managements show an increase of cycle length, correlated to an increase of UO_2 enrichment (close to 5 per cent) and to an increase of core reactivity in the beginning of the cycle. The main neutronic impact of these evolutions is a decrease of the reactivity control systems efficiency (boron and RCCA) due to the hardening of the neutron spectrum (shift towards higher energy levels).

Reactivity control is addressed through different aspects: the reactivity shutdown margin, the safety injection system and the capacity of the make-up system to control the boron concentration (particularly for shutdown situations). It is necessary to verify that the control system is compatible or

adaptable with the new core management (UO_2 enrichment, burn-up, boron concentration). Especially, calculations have to be made for different situations of accidental transients which are to take into account in the safety studies (steam pipe breaks, LOCA).

The typical modifications which can be considered acceptable by utilities (at low costs) are:

- RCCA number increase to the limit compatible with the vessel design;
- RCCA design (adoption of more efficient absorbers);
- safety injection boron concentration increase;
- make-up boron concentration increase and tank levels setpoints modifications;
- boron concentration increase of the refuelling water storage.

Other modifications can improve significantly the safety margins by they are more expensive, the main typical are:

- modification of tanks (volume increase) in the safety systems;
- use of enriched boron (10B concentration increase to 30 per cent or 40 per cent).

Reinforcement of safety demonstration

The raising of the authorised burnup limit makes it necessary to demonstrate the feasibility of all the safety rules and hypotheses developed for this new burnup range and to validate the safety criteria. In fact, the different criteria used in safety analyses are based on experimental data and operational feedback belonging to a burnup range that is well short of the burnup levels targeted.

New experimental programmes should therefore be designed to validate a wider burnup range. The RIA tests being carried out in France (CABRI) to validate the safety criteria observed during rod ejection and the irradiated cladding quench tests regarding LOCA criteria are necessary steps.

The Safety Authority requires that a significant increase in burnup or a significant change in industrially-manufactured cladding materials should be subject to such tests.

Conclusions

The increase of uranium and plutonium fuel burnups is the main line of the fuel development programme as it will lead to reach more cost-effective management of reactor operation. The trend towards higher burnups, however, has led the Safety Authority to demand the validation of the entire safety apparatus – a somewhat laborious approach which often involves carrying out expensive experimental tests programme that affect the time schedule.

The recycling by most PWRs of material arising from reprocessing presupposes the achievement of the equivalence with UO_2 in the way to reinforce its competitiveness. As regards MOX, the fact that it is in its early stages of development gives reason to believe that, when high burnup feedback will be available, it has the potential to achieve the same performance as its predecessor, UO_2.

SPENT FUEL TREATMENT AND WASTE MINIMISATION

P. Parkes and J.-G. Devezeaux
British Nuclear Fuels plc Cogema
United Kingdom France

Abstract

Social, economic, and political factors which may influence a scenario of 1 GWe of installed nuclear capacity, and issues such as uranium supply, were dealt with in the previous session. Innovative alternatives to reactor concepts, fuel forms, and fuel treatments will be dealt with in the next session. In this paper we will discuss the opportunities and challenges from evolution of the current Purex technology, the related issues of uranium and Mox recycle, and effluent and waste treatment.

Scenarios – what, where, and when

In 2020-2030, oxide fuel will be standard, and the bulk of irradiated fuel arising will be from Light Water Reactors due to the long life-spans of capital intensive nuclear facilities[1]. Due to the extremely conservative nature of utilities in the light of high costs to justify changes to operating licences, this fuel is likely to be to similar specification to fuel we see today. Although the trend has favoured higher enrichment to achieve higher performance, average burn-ups are now approaching an optimum economic band for reactors in the 50-60 GWd/t region[2]. Of course, much higher burn-up mean that the proportion of fission products and minor actinides in the fuel will increase, and this causes some operating challenges to existing plants. Safety cases are now in hand to extend operating envelopes for current plants to around 50 GWd/t, and BNFL and Cogema both continue to carry out work to widen the operating envelope of their current commercial reprocessing plants to accept higher burn-ups, if required. Encapsulation processes are also being improved to achieve higher incorporation of minor actinides. Present cladding materials are well characterised and quite capable of these burn-ups. Hence, the irradiated fuel we must manage will be similar to current experience. Thus, the feed materials to the back end pose few challenges to new plants.

Current specifications for uranium require extremely high decontamination factors so that it can be re-enriched and re-fabricated alongside virgin uranium. Alternative re-enrichment processes under development, such as by the use of lasers, may reduce the chemical purity requirements and so reduce processing and associated secondary effluents.

Thermal MOX specifications, and hence process feeds and products separated by reprocessing, have been kept high to keep MOX fuel very similar to uranium fuel and so minimise re-licensing issues. (This has not been an issue for Fast Reactor development). As the thermal MOX market matures, it may be possible to relax these specifications for new build plant if these ease up-stream processing requirements and reduce direct or indirect costs.

Where practical, generating capacity is sited close to where it is to be utilised. For the future, an holistic approach involving "reactor parks" is being considered to:

- closely integrate the fuel cycle;
- reduce construction costs;
- minimise potential environmental impact;
- ease licensing; and
- reduce transport requirements.

If such parks came about then they would be natural homes for reprocessing capacity and any associated re-fabrication plants. Any fast reactor capacity would sensibly be co-located with recycle facilities.

Reprocessing of 10 000 t/a of irradiated fuel would reduce high level waste volumes, containing 99% of the activity, to around 3 000 m^3 using current technology. Whether for direct disposal of fuel or for disposal of smaller quantities of waste conditioned by reprocessing, final repositories for high level waste still need to be established. As some countries do not contain the best geology and geography for a large deep repository with acceptable economics, a world-view points to consideration of international repositories. Again, these would be drivers for siting reprocessing capacity to minimise transport requirements.

As time in reactor and cooling ponds is many years, the challenges of multiple recycle on a large scale is decades away and needs not be faced by current facilities. Large scale recycle, however, will eventually lead to large volumes of irradiated Mox which can justify dedicated facilities to recycle to fast and thermal reactors. Alternative enrichment technologies under development may further simplify multiple recycle.

Scale of the challenge

At the present time, there is around 400 GWe of installed civil nuclear capacity, giving rise to around 10 000 t of irradiated fuel per annum, with a cumulative total of 180 000 t of irradiated fuel discharged to date[3]. Of this quantity, around 60 000 t have been processed[4], with the rest in interim storage awaiting future processing or direct disposal to a repository. The bulk of this fuel has been processed in commercial facilities in England and France, with significant processing carried out in the USA before 1972.

One TWe of nuclear capacity is equivalent to 1 000 modern PWRs. At current fuel burn-ups, this generation capacity would consume and give rise to up to 20×1 000 t/a of fuel. Even when Japan's commercial reprocessing plant comes on stream at Rockasho-Mura, however, current recycling

capacity for oxide fuel will be sufficient for less than 4 000 t/a. Thus, this scenario would require a five fold increase in world reprocessing capacity to attain full thermal recycle and associated reductions in high level waste volumes. Such investment would increase the evolutionary rate of the technology and provide additional reductions in unit costs from series building, as well as providing a further driver to minimise environmental impact from operations.

Evolution of Purex reprocessing

The following section covers developments in reprocessing using Purex technology under the general headings of:

- safety and the environment;
- management of plutonium;
- management of the energy resource;
- technology; and
- scale and economics.

Safety and the environment

The challenges of reducing radioactive and non-radio-active environmental impact must be met to remain a public spirited industry. If the number of nuclear facilities is to rise then the constraints on individual facilities can be expected to become more stringent to limit the cumulative perceived environmental impact.

When current generation facilities were designed they had the option of massive dilution to mitigate the radio-active impact of aerial and liquid effluents. Thus, the effluent volume created by secondary and tertiary wash and scrub cycles to maximise decontamination factors in the products was not a serious issue (DFs for products of up to 10^8). With increasingly stringent limits on discharge, however, the volumes and complexity of the effluent streams will continue to decrease in next generation facilities so that efficient treatment is not prohibitively expensive with regard to the environmental benefit.

The design philosophy of current facilities involves high number of air changes, partly to facilitate man-access, and results in massive volumes of air for scrubbing before discharge. Thus, there is a high dilution factor and a practical limit to the decontamination factor which can be achieved. Next generation facilities, which may not have the same constraints of man-access for example, will re-visit the ventilation aspect of the basis of design and process selection criteria to dramatically reduce the air volumes going through the plant and hence increase practical decontamination factors.

Reprocessing, conditioning, and recycle allow conversion of waste residues to specific requirements of society for disposal to permanent repositories. Vitrification processes in current facilities meet the challenge of providing a safe mechanism for immobilisation and indefinite storage of high level waste. Current developments are investigating matrices and supporting processes to increase the incorporation factor and heat loading of the waste form in order to reduce the volume arising, as well as the environmental impact for generations in the distant future.

Management of plutonium

Conventional reprocessing produces separated plutonium, as it is the route to utilising the vast resources of ^{238}U. A recent study by the NEA showed that there were no technical challenges to the safe management of plutonium in OECD countries. Non the less, it is responsible to ensure that stocks of separated plutonium do not vastly exceed requirements for the fabrication of MOX fuel to extend the availability of valuable virgin uranium resources[5].

Management of the energy resource

As installed nuclear capacity increases and pushes up the consumption rate of uranium, it is likely to result in increases in the price of uranium and hence favour recycle of not only uranium but also plutonium. Recycle of the fissile uranium and plutonium from this fuel to similar thermal reactors could be used to fabricate around a quarter of the volume of fresh fuel, thus avoiding mining more than 50 000 t of virgin uranium per annum under this scenario[6,7]. This is equivalent to about 300 million tones of oil, i.e. about half of Europe's present oil consumption.

Again, this would call for an order of magnitude increase in the scale of thermal MOX fabrication and enable reductions in unit costs similar to those described above.

At some point in this time period, the changing economics would require a review of current recycle strategy, i.e. the balance between breeding and burning of plutonium and the role of fast reactors. Fast reactors are currently being viewed, not for their ability to be self sustaining by breeding plutonium, but for their ability to burn minor actinides and even plutonium and uranium isotopes which can be problematic after several cycles through thermal reactors. These long-lived actinide species have the major potential environmental impact when consigned to a repository. Thus, balanced use of fast reactors enables prolonged thermal recycling as well as minimising the environmental impact from final repositories.

Integrating plutonium in a cycle involving fast reactors requires low product purification and affords maximum safeguardability. Although any breeding blanket produces fresh "clean" plutonium, this is ideal for furnishing thermal Mox fuel to further reduce the need for virgin uranium.

Technology

As a general rule, the capital costs of a reprocessing facility make up more than half the lifetime costs and constitute most of the financial risk. Thus, like the rest of the chemical industry, companies carrying out commercial reprocessing have been making significant investments from profits into breaking away from facilities which have low throughputs compared to building volumes. A number of specific development topics are common in the industry, such as: single (solvent) cycle flowsheets, salt free flowsheets, and co-processing. In addition, there are a number of related themes in the evolution of Purex technology[8]:

- adoption of an holistic (or global) approach, i.e. considering up and down stream implications at the out-set, as against, for example, 'end of pipe solutions' for effluent treatment;
- adoption of international standards for waste forms;
- a reduction in the number and types of waste forms produced;
- a move to intensified process and a reduction in secondary waste streams;
- a move from stochastic to deterministic safety cases;
- a move to real time measurement and control;

- a move from batch processes to continuous processes; and
- a move away from manual operations.

The points give a cascading effect which, when successfully implemented, can eliminate a large number of secondary process control vessels which can result in significant reductions in the foot-print, and hence cost, of a facility. Reductions in the number of secondary effluent and waste streams reduces the number of process stages and associated service plants and hence further reduces capital and operating costs. The last point, aside from reductions in operating costs, removes many of the man-access constraints of the process environment and so can have dramatic effects on capital costs.

The evolution of Purex facilities is thus likely to have a significant impact on back-end fuel cycle costs.

Economics of scale

Despite increased requirements due to more stringent regulatory requirements, OECD/NEA reports on the economics of the nuclear fuel cycle show overall reduction in unit costs with new generation facilities[9]. The cost of reprocessing services is dominated by high fixed costs and low marginal costs, hence high volumes lead to lower unit costs. Thus a facility to treat 80 000 t of oxide fuel from US utilities, presently earmarked for direct disposal, would have quite a different cost structure from one serving a much smaller home market. With significant scale effects in conjunction with technical progress, the unit costs for next generation reprocessing could be reduced by more than a factor of 1.5, depending on the size the plant and the specification for separated products.

Conclusions

- There are no technical obstacles in sustaining this nuclear capacity at the back-end through expansion of Purex technology.
- Unit costs could reduce by more than a factor of 1.5 through deployment of evolutionary technology, as well as from economies of scale and series building.
- Environmental impact is negligible today and would not grow in proportion to installed capacity, as next generation facilities would be geared to these challenges.
- Plutonium stocks can be actively managed through a number of options including safeguarding or recycle, if required by specific customers.
- True sustainability at the front-end would require maximising resources through a recycle strategy.
- There are no technical obstacles to supporting back-end MOX or Fast Reactor deployment.
- Safeguardability can be further improved, if required, through complimentary reactor and fuel cycle combinations.
- Recycle strategy, i.e. extent of deployment of MOX and Fast Reactor, would be driven by economics from U availabity/price and by policy for waste conditioning.
- Reprocessing allows a large set of possible adaptations for conditioning of waste, if needed for specific repository scenarios, including: removal of specific nuclides, their chemical form, and the encapsulating matrix.

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2. Paper by Kevin Hesketh – optimum ca 55 GWd/t.

3. NEA data from “Nuclear Fuel Cycle and Strategies: Adjusting to New Realities”.

4. Aside from military processing, which has been significant and represents several tens of thousands of tonnes in the USA, Russia, China, France and the United Kingdom.

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HIGH-LEVEL WASTE INCINERATION AND Pu MANAGEMENT BY RECYCLING IN LMFRs

Toshio Wakabayashi and Kiyoshi Ono

Oarai Engineering Center
Power Reactor and Nuclear Fuel Development Corporation, Japan

Abstract

Systematic studies were implemented to investigate the feasibility of minor actinide (MA) and long-lived fission product (LLFP) transmutation and Pu burning in liquid metal fast reactors (LMFRs). MA transmutation in a fast reactor core has no serious drawbacks in terms of core performance, provided that the homogeneous loading method can be employed with a small fraction of MA fuel (~5wt per cent). The recycling of MA in a fast reactor is feasible from neutronic and thermal-hydraulic points of view. For FP transmutation, the introduction of target subassemblies using duplex pellets – a moderator annulus surrounding a ^{99}Tc core – gives the maximum transmutation rate of ^{99}Tc in the radial shield region of the fast reactor. Highly enriched MOX fuels and Pu fuels without uranium were considered for Pu burning enhancement. Both burnup reactivity loss and Doppler coefficient are important criteria for highly enriched MOX fuel cores. The introduction of UO_2 in an internal blanket is effective in enhancing the Doppler coefficient, with minor increase in the sodium void reactivity, in non-uranium cores. The fast reactors have an excellent potential for incinerating MA and LLFP and burning Pu effectively. The fast reactors will be able to play an important role in future energy system.

Introduction

One of the distinctive features of a fast reactor is its good neutron economy. Utilising the excess of neutrons enables us to construct flexible cores such that they incinerate minor actinides (MAs) and long lived fission products (LLFP) to reduce radiotoxicity and breed or burn plutonium in consideration of plutonium balance.

Some of the MA nuclides (Np, Am, Cm) contained in residual waste from reprocessing have extremely long-lived radiotoxicity.[1] Means of reducing the radiotoxicity of the MA nuclides are presently under investigation. The MA nuclides could produce useful energy if converted into short-lived fission products by neutron bombardment. From this standpoint, a nuclear reactor provides the obvious means for transmutation of MA nuclides. Among the various nuclear reactors, a fast reactor is considered to have the greatest potential to transmute MA effectively, because of its hard neutron spectrum.[2-6]

The beta-emitting fission products technetium (^{99}Tc, half-life 2.13×10^{5} year) and iodine (^{129}I, half-life 1.57×10^{7} year) are among the important long-lived nuclides in high-level waste, they dominate the beta radiotoxicity for more than a million years. Transmutation of ^{99}Tc and ^{129}I by neutron capture as a result of irradiation in nuclear reactors will yield the stable isotopes ^{100}Ru and ^{130}Xe, respectively. However, due to the small neutron cross sections, the transmutation efficiency in LWRs is low. Moderated subassemblies in fast reactors are more appropriate devices for the transmutation of the fission products.[6-8]

This paper describes the feasibility of MA and LLFP transmutation in fast reactor cores.

There has been increasing focus on the research and development work necessary for the utilisation of the excellent Pu burning characteristics of fast reactor cores. Studies on Pu burner fast reactor cores have been performed that show the flexibility of plutonium utilising characteristics of fast reactors.[9-10]

The following three approaches to burning plutonium efficiently in a fast reactor are considered:

- enhancement of neutron leakage (high pu enrichment mox core);
- introduction of neutron absorption material (high pu enrichment mox core);
- core without uranium.

This paper also describes Pu burning characteristics in a fast reactor. Several series of analyses were performed by changing various parameters including fuel pin specifications, smear density, core height, Pu vector, types of inert matrix without U. The effects of some options to improve the core characteristics are also discussed.

MA Transmutation

Systematic parameter survey calculations were implemented to investigate the basic characteristics of MA transmutation based on a 1 000 MWe-class large FBR core with mixed oxide fuel. The design parameters are shown in Table 1.

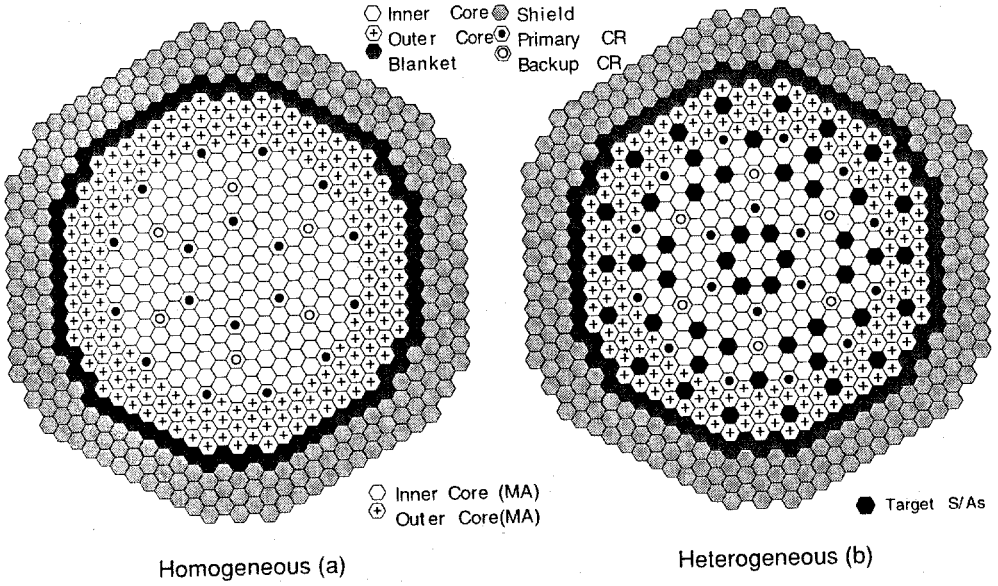
Table 1. **Main design parameter of the 1 000 MWe LMFBR**

Reactor thermal power (MWth)	2 517
Reactor electric power (MWe)	1 000
Cycle length (months)	15
Core Concept	Two-region homogeneous
Core diameter/core height (m)	3.68/1.00
Thickness of axial blanket (cm)	20
Driver fuel Fuel composition	MOX
Pu enrichment (wt%) (inner/outer)	15.4/18.6
Pin diameter (mm)	7.22
Refuelling batches	3
Blanket fuel Fuel composition	UO_2
Uranium isotopic ratio (235/238)	0.3/99.7
Pin diameter (mm)	12.4
Refuelling batches	4
Coolant	Sodium
Coolant temperature (outlet/inlet) (°C)	530/375

Study on MA loading method

Since MA loading considerably affects not only core characteristics but also fuel material properties, it is necessary to investigate MA loading methods taking into account this influence upon core characteristics and fuel material properties. Possible MA loading methods (homogeneous, heterogeneous, blanket, etc.) were investigated for fast reactor cores with no special design adaptation for the MA loading. The MA fuel is dispersed uniformly throughout the core in the homogeneous method, as in shown Figure 1(a). In the heterogeneous method, a small number of subassemblies with concentrated MA fuel (target subassemblies) are loaded into the core, as in shown Figure 1(b).

Figure 1. **MA loading method**



The comparison of core performance for various MA loading methods is shown in Table 2. The MA transmutation in a fast reactor core has no serious drawbacks in terms of core performance, provided that the homogeneous loading method can be employed with a small ratio of MA to fuel (~5wt per cent). Since a 1 000 MWe-class LWR produces ~26 kg of MA per year, a fast reactor with 5 per cent wt-MA loading can transmute the MA produced by six LWRs.

Table 2. **Comparison of core performance for various MA loading methods**

Item	Reference (No MA)	Homogeneous loading	Heterogeneous loading	Homogeneous loading
MA and RE loaded in the core region	–	Np, Am, Cm: 5% RE:0%	Np, Am, Cm: 9% RE: 0% (Number of target S/As: 39)	Np, Am, Cm: 5% RE: 10 %
Matrix of target	–	–	UO_2	–
Core height (cm)	100	100	100	100
Cycle length (days)	456	456	456	456
Number of batches	3	3	3	3
Pu enrichment (inner core /outer core) (wt%)	15.4/18.6	16.6/20.1	15.4/18.6	20.0/24.2
B.U. reactivity loss (%δk/kk')	3.31	2.12	1.83	3.71
Max. linear heat rate (driver/target) (W/cm)	420	407	439/309	413
Void reactivity	1.0	1.3(1)	1.3(1)	1.4(1)
Doppler coef.	1.0	0.6(1)	0.7(1)	0.5(1)
MA transmutation Amount (kg/cycle) Rate (%/cycle)	– –	172 10.9	186 11.3	164 10.3

(1) Relative values.

The heterogeneous MA loading method can be made feasible by optimising the fuel design, loading pattern and the coolant flow of the MA loaded fuel subassemblies. The reduction of the fuel pin diameter and the Pu enrichment is essential to reduce the power of MA loaded fuel in the heterogeneous MA loading method.

The MA loading in the blanket region causes no problems from the viewpoint of core performance. Minor actinides are transmuted at a rate of 6 per cent per cycle in the axial and radial blanket regions.

Selection of fuel material for MA transmutation

Different types of inert matrices, instead of uranium, for the heterogeneous MA loading method have been investigated; they avoid the build-up of higher actinides via ^{238}U and achieve a high MA transmutation rate. Inert matrices of Al_2O_3 and CeO_2 were examined in this study. The transmutation rate and the isotopic composition of MA discharged from target pins are shown in Table 3. The MA transmutation rate of the target subassembly using inert matrices is larger than that of the target subassembly using UO_2: the inert matrices in the target subassembly effectively increase the MA transmutation rate.

Table 3. **Isotopic composition of MA discharged from the target fuel using various matrices***

Matrix of target	^{237}Np	^{241}Am	^{243}Am	^{244}Cm	^{235}Cm	Transmutation ratio(%)
Al_2O_3 (loading)	0.0	1.0	0.52	0.0	0.0	–
Al_2O_3 (discharged)	0.0	0.4	0.23	0.22	0.03	42
CeO_2 (loading)	0.0	1.0	0.52	0.0	0.0	–
CeO_2 (discharged)	0.0	0.52	0.30	0.18	0.02	34
UO_2 (loading)	0.0	1.0	0.52	0.0	0.0	
UO_2 (discharged)	0.01	0.61	0.34	0.14	0.01	28

* All values normalised at the amount of ^{241}Am

Study on the permissible rare earth (RE) level in homogeneously loaded MA

Systematic parameter survey calculations were performed to investigate the basic characteristics of a fast reactor core loaded homogeneously with MA which contains RE, and also to establish a MA and RE loading method which has no serious influence on the core design. The homogeneous loading of MA and RE has no serious effects on the reactor core performance, provided that the amounts of MA and RE in the fuel are less than 5 and 10wt per cent respectively, as shown in Table 4. In the case of adding Am, Cm and RE in the radial blanket region, it is possible, from the viewpoint of core performance, to insert ~50wt per cent of Am and Cm, and ~50wt per cent of RE in the target assemblies.

Table 4. **Effect of RE content on core performance (homogeneous loading method)**

Item	Reference (RE=0%)	RE=10%	RE=30%
MA (wt%)	5	5	5
Pu enrichment (inner core /outer core) (wt%)	16.6/20.1	20.0/24.2	29.2/35.4
B.U. reactivity loss (%δk/kk')	2.12	3.71	6.40
Void reactivity	1.0 (reference)	1.02 (relative value)	–
Doppler coef.	1.0 (reference)	0.87 (relative value)	–
MA transmutation (%/cycle)	10.9	10.3	9.7

Effect of MA recycling on core characteristics and fuel cycle system.

The effects, on the core characteristics and fuel cycle system, of MA recycling in the homogeneous loading method, were evaluated. The absolute value of the Doppler coefficient is increased by MA recycling, the value at the 8th recycle is ~14 per cent larger in comparison with that for the initial core, as is shown in Table 5. This is caused by the reduction in Pu enrichment with MA recycling, this increases the resonance absorption of ^{238}U. Sodium void reactivity decreases with MA recycling and the value at the 8th recycle is ~7 per cent smaller than that for the initial core. The recycling of MA in a fast reactor is feasible from neutronic and thermal-hydraulic points of view. However, the multi-recycled Np fraction is significantly depleted compared to the unirradiated feed, and the fraction of Cm is greatly increased because of neutron capture in Am. The accumulation of Cm as a result of the MA recycling will bring about some problems concerning fuel handling and reprocessing, because of an increase in both the decay heat and the neutron emission rate from ^{244}Cm.

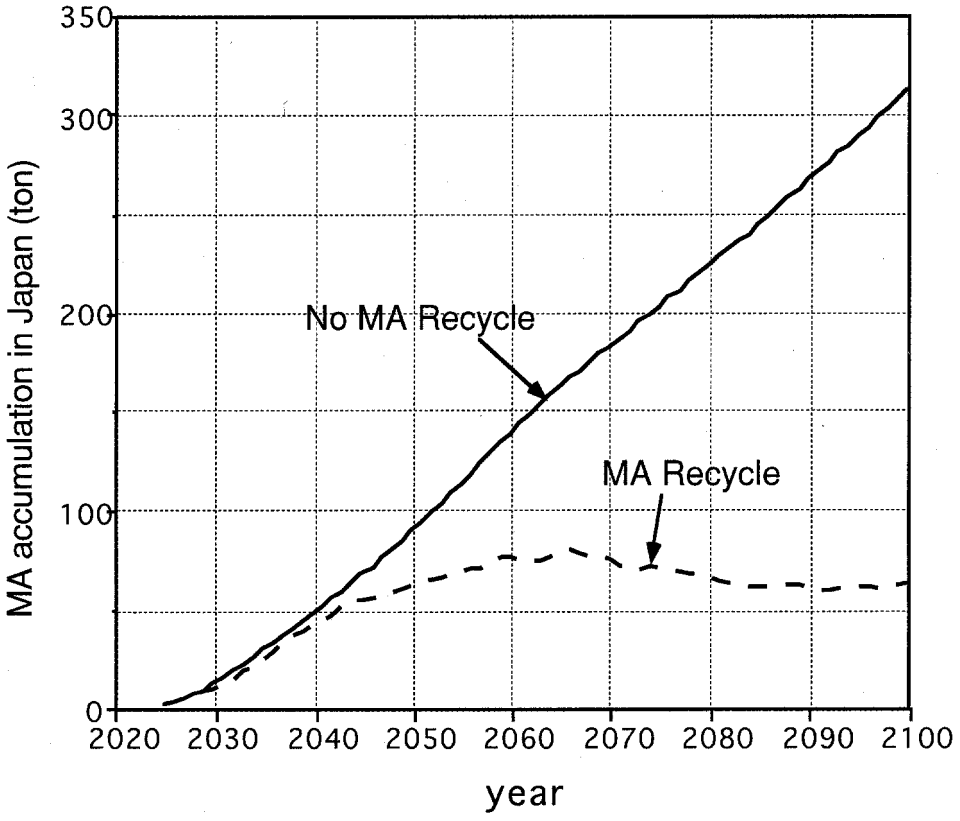
Table 5. **Effect of MA recycling on core performance (homogeneous loading method)**

Item	Reference (Initial core)	Fourth recycle	Eighth recycle
MA (wt%)	5	5	5
Pu enrichment (inner core /outer core) (wt%)	17.8/21.6	17.6 /21.3	16.9 /20.4
B.U. reactivity loss (%δk/kk')	1.6	0.4	0.5
Void reactivity	1.0 (reference)	0.96 (relative value)	0.93 (relative value)
Doppler coef.	1.0 (reference)	1.08 (relative value)	1.14 (relative value)
MA transmutation (%/cycle)	10.3	10.5	10.1

Effect of the reduction of MA inventory

The MA mass balance was analysed according to the predicted nuclear energy production in Japan. Plutonium and MAs are recovered from the LWR and Pu-thermal reactors, recovered Pu and MAs are multiply recycled in fast reactors. Nuclear power generation is assumed to increase to 1 000 MWe/y, with the introduction of commercial fast reactors starting in the year 2030. New reactors are assumed to be totally FBR, and all spent fuel discharged from LWR and Pu-thermal reactors is assumed to be reprocessed. The total MAs transferred into the high level waste are calculated to be 310 tons from LWR, Pu-thermal LWR and FBR without recycling. In the case of recycling MAs into LMFRs after the year 2030, the MAs remaining in the fuel cycle in the year 2100 is reduced to about 60 tons, 80 per cent less than without recycling, as shown in Figure 2.

Figure 2. **Effect of transmutation in reducing accumulation MA**



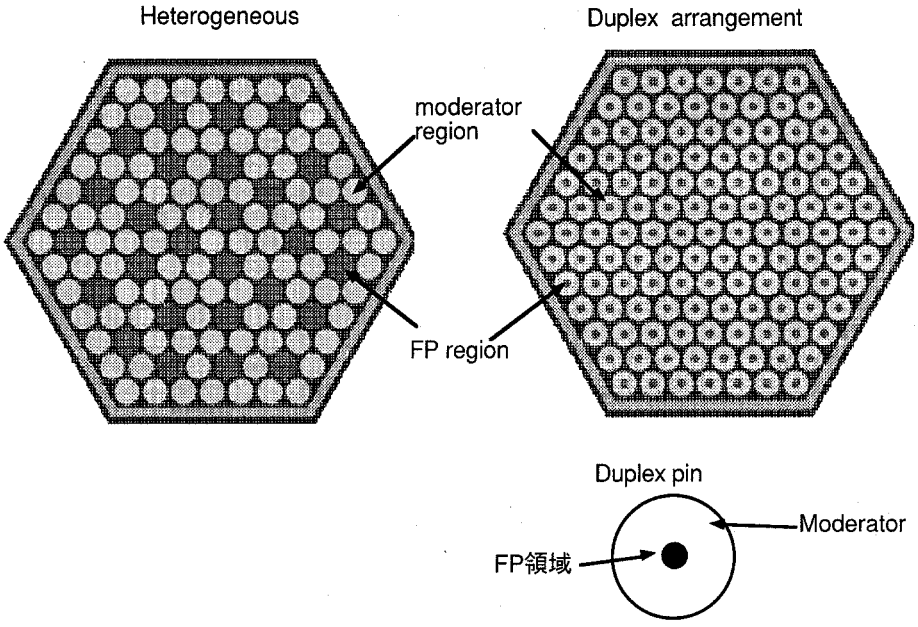
LLFP Transmutation

FP transmutation method in a fast reactor

To calculate the transmutation rate for ^{99}Tc in a neutron flux spectrum it is insufficient to account for the thermal neutron capture only, the epithermal part of the neutron spectrum also has a contribution. There is a large resonance peak at 5.6eV and a series of minor resonances between 10 and 100eV. This suggests that a neutron spectrum where there is a higher flux in the resonance region than in the thermal region is advantageous in order to increase the transmutation rate of ^{99}Tc. This is because such a spectrum helps to suppress absorption by structural materials. Therefore, the appropriate loading mass of moderator depends upon its moderating power.

A moderated target subassembly was used for FP transmutation. The subassembly consists of moderator pins and FP target pins distributed between the moderator pins. The moderated target subassemblies were loaded in the radial shield region of the fast reactor. A new concept of duplex pellets was also examined: a moderator annulus surrounding a ^{99}Tc core, as shown in Figure 3, adopted to get a better ^{99}Tc transmutation performance.

Figure 3. **Configuration of moderated target subassemblies**



FP transmutation performance

Systematic parameter survey calculations were performed to investigate the basic characteristics of FP transmutation in the blanket region of a fast reactor. The arrangement of the moderator and the target pins in the subassembly, the moderator material and the volume ratio of target to moderator were selected as parameters. The results of the calculations are shown in Table 6. The transmutation rate of ^{99}Tc in the new target subassembly is higher than that in the subassembly consisting of separate $ZrH_{1.7}$ moderator pins and ^{99}Tc target pins, as shown in Figure 3. A maximum ^{99}Tc transmutation rate of about 10 per cent/year was obtained by using the new target subassembly loaded in the blanket region of the fast reactor. The new target subassembly can achieve an optimum transmutation performance by adjusting the volume ratio of $ZrH_{1.7}$ to ^{99}Tc in the duplex pellet.

The effects on main core characteristics of loading target subassemblies were also analysed. It was found that the power density of the core fuel adjacent to the target is rather high and is about the same as the maximum in the core. However, the power spike is much mitigated compared to the case of loading target subassemblies in the core region.

Table 6. **Results of ^{99}Tc transmutation performance parameter survey**

Loading method of FP pins	Number of pins in subassembly	Number of FP pins	Radius of FP pin	Transmuted amount (kg/y)	Transmutation ratio (%)
Heterogeneous	127	37	0.5	41.1	1.8
Heterogeneous	127	22	0.5	27.2	2.5
Duplex	127	127	0.2	38.1	3.5
Duplex	127	127	0.063	10.8	9.8
Duplex	217	217	0.2	46.7	2.5
Duplex	217	217	0.063	17.1	9.1

Several calculations were performed to determine the ^{129}I transmutation performance. ^{129}I was loaded as NaI. The isotopic concentration of ^{129}I was 76.5 per cent and the remainder ^{127}I. The transmutation rate of ^{129}I was 5.2 per cent and the transmuted amount was 18 kg in a year. The amount of ^{129}I produced by a 1 000 MWe class PWR is about 5.0 kg, so the transmuted amount of ^{129}I was equal to the output from 3 PWRs.

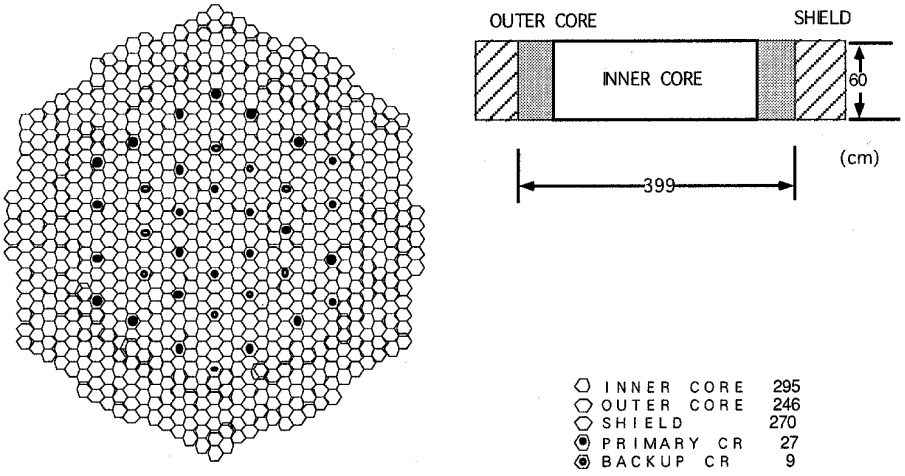
Plutonium Burning

High Pu Enrichment MOX Core

Core specification

A 600 MW electric core (1 600 MW thermal) was selected as the reference in this study. The core is a two-region homogeneous type of 60-cm height; it has a pancake shape to enhance neutron leakage. The core diameter is ~4m, to keep a half-a-year operation period as a minimum requirement for a commercial reactor. The core configuration of the 600 MWe-class Pu burner core is shown in Figure 4.

Figure 4. **Configuration of the plutonium burner core (600 MWe, MOX fuel)**



Study items are the general core characteristics of the Pu burner, the effect of Pu vector and the effect of the heterogeneous introduction of B_4C absorbers in the core.

General core characteristics of Pu Burner

A series of calculations were performed to investigate the Pu consumption rate of burner cores, changing fuel pin specification, smear density, core height, etc.

The core characteristics of the Pu burner are summarised in Table 7, which also shows for comparison those of a MOX-type breeder core. The Pu consumption rate of the reference core is about 75 kg/TWhe. The burnup reactivity loss is increased by ~40 per cent compared with the breeder core. However, a good reactivity balance is confirmed, with neither the use of removable absorber subassemblies nor an increase in the number of control rods. In order to decrease the fuel volume an increase in coolant volume was adopted; since this reduces sodium void reactivity: the increase of coolant volume makes the leakage effect larger when in-core sodium is replaced with void. The sodium void reactivity is decreased by approximately 1.2 per cent δk/kk'. The Doppler coefficient is decreased by around a half compared with the breeder core. It was concluded that the 600 MWe-class Pu burner core is feasible from the neutronic viewpoint.

Table 7. **Core characteristics of Pu burner (600 MWe)**

Core type	Pu Burner(1) (Reference)	Degraded Pu(2)	High Puf(3)	B_4C S/A(4)	Breeder
Core height (cm)	60	60	60	60	100
Cycle length (days)	183	183	183	183	456
Number of batches	4	4	4	4	3
Pu enrichment [inner core /outer core (wt%)]	37.5/45.0	37.5/45.0	37.5/45.0	37.5/45.0	15.8/22.2
Pu consumption (kg/TWhe)	74.7	78.0	72.4	78.7	–
Puf consumption (kg/TWhe)	76.5	58.8	95.6	78.4	–
Max. linear heat rate (W/cm)	331	291	289	408	412
Conversion ratio	0.38	0.49	0.26	0.34	1.21
B.U. reactivity loss (%δk/kk')	4.7	4.0	5.4	3.8	3.0
Void reactivity (%δk/kk')	+1.0	+2.3	-0.2	+1.4	+2.3
Doppler coef. (x10-3Tdk/dT)	-4.3	-4.1	-4.1	-2.7	-7.9
C/R rod worth (Requirement) (%δk/kk')	7.3 (6.7)	–	–	5.5 (5.5)	7.0 (6.0)

Pu Isotopic Composition [^{238}Pu/^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu; (w/o)].
(1) Reference: (1.8 /58.8 /22.5 /11.2 / 5.6).
(2) Degraded = (49.1/30.0/0.08/15.5/0.05/5.0/0.3).
(3) High Puf: (0.0 /94.0 /6.0 /0.0 / 0.0).
(4) Enrichment of ^{10}B: 90wt%.

Effects of Pu isotopic composition

Table 7 also shows the effects of the Pu isotopic composition on core performance. The burnup reactivity loss in the core with the degraded Pu is decreased by ~15 per cent compared with the reference core. On the other hand, the burnup reactivity loss in the core with the high Puf content[11] is increased by ~15 per cent. The sodium void reactivity in the core with the degraded Pu is increased by ~1.3δk/kk' compared with the reference core. The sodium void reactivity in the core with the high Puf is nearly zero. The Doppler coefficients of the cores having the degraded Pu and the high Puf are almost the same as the value of the reference core. The Pu burner core has an excellent potential for burning plutonium with various kinds of Pu vector.

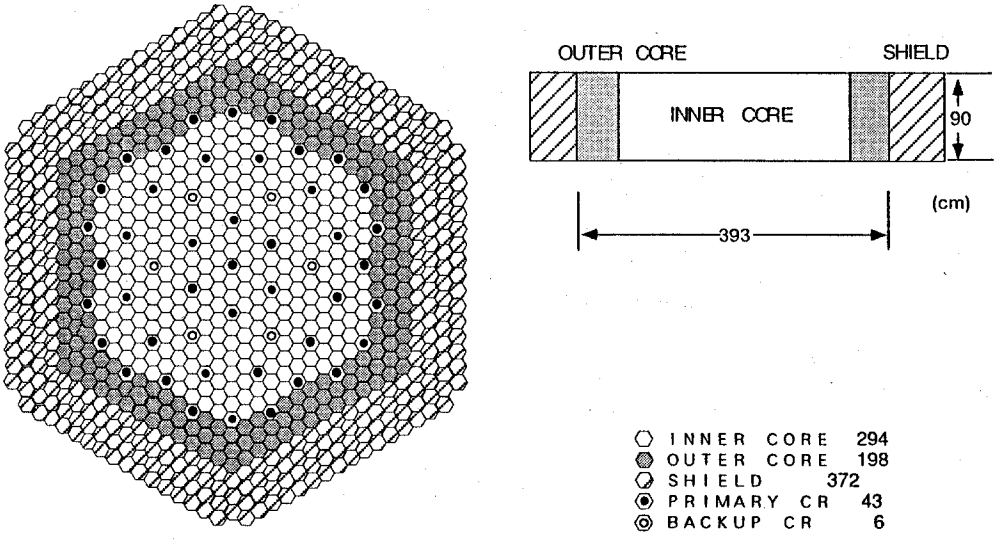
Effects of heterogeneous introduction of B_4C absorbers

In order to reduce the burnup reactivity loss, B_4C with enriched ^{10}B as burnable poison was introduced heterogeneously in the core region of the MOX Pu burner. The effects are summarised in Table 7. The burnup reactivity loss is decreased by ~20 per cent. The sodium void reactivity is increased by approximately 0.4δk/kk'. The Doppler coefficient is decreased by ~37 per cent. The presence of the ^{10}B absorber reduces the primary control rod worth by 25 per cent, it remains just adequate to requirement.

Non-Uranium Core

A core of 600 MW electric was selected as the reference in this study, as shown in Figure 5. The core is a two-region homogeneous type of 90 cm height. Core characteristics are evaluated for non-uranium cores produced by replacing UO_2 with ceramic, either Al_2O_3 or BeO. In each case the Pu fraction in the fuel is adjusted to assure criticality during operation. Study items were the general characteristics of the reference core, the effect of heterogeneous B_4C absorber introduction, the effect of the homogeneous UO_2 introduction and the effect of UO_2 introduction in an internal axial blanket.

Figure 5. **Configuration of the core without uranium (600 MWe, PuO_2-Al_2O_3)**



Core characteristics of non-uranium core

The core characteristics of non-uranium cores are summarised in Table 8, which also shows for comparison those of the reference MOX-type Pu burner core.

Table 8. **Core characteristics of non-uranium cores (600 MWe)**

Core type	PuO_2/Al_20_3	PuO_2/BeO	B_4C S/A(1)	Pu Burner (MOX)
Core height (cm)	90	90	60	60
Cycle length (days)	270	270	270	183
Number of batches	6	6	6	4
Pu enrichment [inner core/outer core (wt%)]	36.9/43.1	43.2/49.5	65.0/48.1	37.5/45.0
Pu consumption (kg/TWhe)	113.8	114.0	116	74.7
Puf consumption (kg/TWhe)	115.8	117.5	117	76.5
Max. linear heat rate (W/cm)	317	350	311	331
Conversion ratio	0.16	0.21	0.12	0.38
B.U. reactivity loss (%δk/kk')	8.9	9.5	6.2	4.7
Void reactivity (%δk/kk')	-0.26	+0.03	-0.45	+1.0
Doppler coef. ($x10^{-3}$Tdk/dT)	-4.7	-8.8	-2.2	-4.3
C/R rod worth (Requirement) (%δk/kk')	14.0 (12.8)	13.8 (13.7)	9.1 (9.0)	7.3 (6.7)

(1) Enrichment of ^{10}B: Natural

The Pu consumption rates of the non-uranium cores are almost double that of the MOX fuelled core. The value of 110 kg/TWhe is close to the theoretical maximum. The burnup reactivity loss is doubled to 9.5 per cent δk/kk', because of the lack of ^{239}Pu production. However, a good reactivity balance is confirmed, with neither the use of removable absorber sub-assemblies nor an increase in the number of control rods.

The sodium void reactivity shows a marked decrease and reaches negative values: this is very favourable from a viewpoint of reactor safety. Component analysis using exact perturbation theory was used: it determined that the dominant change occurs in the scattering term, which is caused by the spectrum softening. The effects of the spectrum change were also seen in the absorption and leakage terms. Despite the absence of ^{238}U, the Doppler coefficients of the non-uranium cores remain negative. The Doppler coefficient for the PuO_2/Al_2O_3 core is close to that of the MOX Pu burner core. The Doppler coefficient for PuO_2/BeO core is larger, close to that of the conventional MOX-type FBR cores. The main contributions to the Doppler coefficient are made by ^{240}Pu and iron.

Effects of heterogeneous introduction of B_4C absorbers

In order to reduce the burnup reactivity loss, B_4C with natural ^{10}B content was introduced heterogeneously in the core region of the non-uranium Pu burner. The effects are summarised in Table 8. The burnup reactivity loss is decreased by around 2.7 per cent δk/kk', the already negative sodium void reactivity is decreased by approximately 0.2 per cent δk/kk', and the Doppler coefficient is decreased by around a half compared with the reference core. These changes are caused by the spectral hardening due to neutron absorption in ^{10}B.

Effect of homogeneous introduction of UO_2

It might be possible that a comparatively small amount of UO_2 can compensate for the unfavourable characteristics of the non-uranium core without serious influence on the Pu burning performance. The effects of the homogeneous introduction of 30 per cent UO_2 is summarised in Table 9. The burnup reactivity loss is improved by ~16 per cent, and the Doppler coefficient by ~20 per cent. On the other hand, the Pu consumption rate is decreased by ~16 per cent, and the sodium void reactivity is increased by ~0.6 per cent δk/kk'.

Table 9. **Effect of UO_2 introduction on characteristics of non-uranium cores (600 MWe)**

Core type	UO_2 introduction (homogeneous)	PuO_2/BeO	Reference (PuO_2/Al_20_3)
Core height (cm)	90	90(IB=11)	90
Cycle length (days)	270	270	270
Number of batches	6	6	6
Pu enrichment [inner core /outer core (wt%)]	37.9/43.9	41.3	36.9/43.1
Pu consumption (kg/TWhe)	96	98	113.8
Puf consumption (kg/TWhe)	100	102	115.8
Max. linear heat rate (W/cm)	298	318	317
Conversion ratio	0.27	0.26	0.16
B.U. reactivity loss (%δk/kk')	7.5	7.0	8.9
Void reactivity (%δk/kk')	+0.32	-0.02	-0.26
Doppler coef. ($x10^{-3}$Tdk/dT)	-5.7	-6.3	-4.7
C/R rod worth (Requirement) (%δk/kk')	11.3 (11.1)	11.5 (10.5)	14.0 (12.8)

Effects of introduction of UO_2 internal axial blanket

In the conventional MOX-FBR design, an axially heterogeneous core is considered as one concept to improve the core performance. A similar concept may be effective for the Pu burner, because a minor amount of ^{238}U can be located exclusively in the region of greatest importance. The core configuration of the non-uranium core with an axial blanket is shown in Figure 6; this is a one-zone core with an internal blanket of 11 cm height. The effects are summarised in Table 9, alongside the heterogeneous UO_2 core. Although the Pu consumption rate is decreased by ~14 per cent, the burnup reactivity loss shows an improvement of ~24 per cent. However, the primary rod worth and margin to requirements are both reduced.

One significant improvement of this option appears in the isothermal Doppler coefficient, which increases by ~34 per cent and is close to that of the conventional MOX-type FBR cores. The increase in the sodium void reactivity is only ~0.2 per cent δk/kk', which is much smaller than that for the homogeneous introduction of UO_2.

It was found that the internal blanket can enhance the Doppler coefficient with minor increase in the sodium void reactivity. Although consideration is still needed for the degradation of the effective Doppler constant, in the actual reactor operation and in transients, because of temperature differences between PuO_2 and UO_2, the adoption of an internal blanket in the non-uranium fuelled cores is worth pursuing further.

Conclusion

The feasibility of the transmutation of minor actinides and long-lived fission products in fast reactors was investigated. The potential of fast reactors for burning Pu was also investigated.

The MA transmutation in a fast reactor core has no serious drawbacks in terms of core performance, provided that the homogeneous loading method can be employed with a small fraction of MA fuel (~5wt per cent). The recycling of MA in a fast reactor is feasible from neutronic and thermal-hydraulic points of view. By recycling MAs in fast reactors after the year 2030, in Japan the MAs remaining in the fuel cycle in the year 2100 can be reduced to ~60 tons, 80 per cent less than without recycling.

For FP transmutation, the introduction of target subassemblies using duplex pellets – a moderator annulus surrounding a ^{99}Tc core – gives the maximum transmutation rate of ^{99}Tc in the radial shield region of the fast reactor.

Highly enriched MOX fuels and Pu fuels without uranium were considered for Pu burning enhancement. It was found that Pu consumption rates essentially depend on Pu enrichment. Both burnup reactivity loss and Doppler coefficient are important criteria for highly enriched MOX fuel cores. The cores without uranium were found to consume the Pu at very large burnup rate, close to the theoretically maximum value of 110-120 kg/TWhe. The introduction of UO_2 in an internal blanket is effective in enhancing the Doppler coefficient, with a minor increase in the sodium void reactivity, in non-uranium cores.

The fast reactors have an excellent potential for incinerating MA and LLFP and burning Pu effectively. The fast reactors will be able to play an important role in future energy system.

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SESSION #3

INNOVATIVE ALTERNATIVES

CHAIRMAN: PROF. J.N. VAN GEEL, THE NETHERLANDS

INNOVATIVE REACTOR CONCEPTS

V.Orlov[1], K. Mikitiouk[2]
[1]RDIPE, [2]RRC "Kurchatov Institute", Moscow, Russian Federation

Abstract

The scenario for development of the large-scale nuclear power with commissioning of new fast reactors is considered. This large-scale nuclear power would allow to replace and stabilize organic fuel utilization and, simultaneously, to meet the growing demands of developing countries in energy. An advanced reactor concept based on a fast lead-cooled reactor is proposed. Inherent properties of this reactor make it possible to solve problems of nuclear safety, economic efficiency, and radwaste handling, as well as to eliminate the possibility to use the power technologies for production of nuclear weapons.

Introduction

In 1940s, developing nuclear weapons, E. Fermi in the USA and A. Leipunski in the USSR came to the understanding that, taking into account the nuclear breeding, resources of nuclear energy are practically inexhaustible and can be used with peaceful aims. They initiated in their countries works on using nuclear energy in the power sector. Somewhat later such activities were started in other countries. This very opportunity to use inexhaustible reserves of nuclear energy became a basis for these countries to undertake the efforts and to meet the risk, when starting the use of the new type of energy source.

E. Fermi early in 1944 made several presentations in which formulated the problems to be solved in this way. Among them are:

- development of economic fast breeding reactors in the U-Pu cycle;
- safety assurance of nuclear power plants (NPPs) and radwaste handling;
- non-proliferation of nuclear weapon.

Actual development of the nuclear power started in 1950s with the use of thermal reactors fuelled by U-235. The types of these reactors were determined by the reactors already developed for production of weapon-grade plutonium (graphite, heavy-water reactors) and for power facilities of nuclear submarines (light-water reactors). It was known that stocks of cheap uranium for these reactors are not sufficient for development of a large-scale nuclear power. Potential resources of cheap natural uranium in the world were estimated as 10^7 tons that are less than oil and gas resources and significantly less than coal stocks. However, it was supposed then that Pu produced in these reactors could be used in future for commissioning of fast reactors with short Pu doubling time and the large-scale power sector (several thousand GW) could be developed already in this century.

High power densities and breeding rates as well as uranium blanket in the first fast reactors were used for reaching short Pu doubling times. These reactors were sodium-cooled and fuelled with best-proven uranium oxide. However, to further increase power densities and breeding rates, fuel types with higher density and thermal conductivity were studied (carbides, mononitrides, U-Pu metal alloys). The first test fast reactors were constructed in the USA and the USSR in 1950s and first NPPs with fast reactors were commissioned in the USSR, France and the Great Britain in 1970-80s. The BN-350 and BN-600 reactors have been successfully operating for about 30 and 20 years, correspondingly. The construction of the BN-800 reactor has been started. However, the first fast reactors were aimed at short Pu doubling times and due to the use of chemically active and low-boiling sodium coolant they proved to be very expensive compared to light-water reactors.

Moreover, plutonium is generated in the blanket of these reactors as well as when the fuel is recycled. These factors increase the risk of the nuclear weapon proliferation, so, the fast reactors did not gain acceptance in the USA and now their development is stopped in the West European countries.

Nuclear Power Development Scenarios

Under conditions when developed countries are saturated with energy, world fuel market is stabilized and there is a strong antinuclear opposition in the society, the fraction of energy produced in the nuclear power based on thermal reactors in the total world electricity production has reached 17%, stabilized on this level and is expected to be decreased in the next decades.

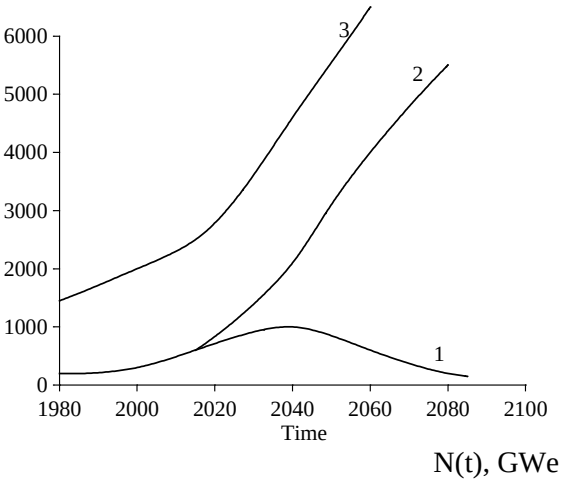
The measures aimed at safety enhancement after large-scale accidents at NPPs (Three-Mile Island and Chernobyl) have decreased accident probabilities by 1-2 orders of magnitude. These measures allowed to continue NPP operation and construction, but increased NPP costs (about 1.5-2 times). This cost increase made nuclear power less or even not competitive with other energy sources. Demand for NPPs has sharply dropped and almost disappeared in the USA and in the West European countries.

At the same time, the developing countries of Asia and other continents with the highest rate of the population growth have to be satisfied with the energy consumption per person that an order of magnitude less than in developed countries. So, these countries are interested in significant increase of energy consumption and stimulate the growth in energy production, including nuclear energy. These countries are naturally starting the development of their nuclear power sector from steps made by the developed countries in this century, i.e., with the use of NPP types that are traditional for the developed countries.

Today, the nuclear power development is limited not by a lack of cheap uranium. However, in the long-term future, development of the nuclear power based on thermal reactors will be restricted by this very factor, if only we do not rely on some speculative estimates of cheap uranium reserves (actual estimates have not significantly changed for the last 30 years) or on the opportunity to use in thermal reactors uranium extracted from sea water (this uranium cannot be competitive with coal). On this basis an approximate scenario for the long-term development of the nuclear power based on thermal reactors can be given by curve 1 on Figure 1.

No plutonium recycle in thermal reactors is supposed in this case, but it could increase nuclear fuel resources only insignificantly (by about 20%). Such a recycle is now implemented in a number of countries with the use, mainly, of the spent fuel reprocessing facilities constructed in France and the Great Britain. These facilities now provide reprocessing of about 25% of spent fuel produced in the world. At the same time, development or construction of new reprocessing facilities based on aqueous methods are not planned because there are no economic ground for it.

Figure 1. **Scenarios for the nuclear power development**



1. thermal reactors; 2. thermal + fast reactors; 3. total world electric power production.

According to our estimates, a MOX fuel assembly for LWR costs 3-4 times higher than uranium one and this economic disadvantage will still exist in future even if natural uranium cost doubles or reprocessing cost halves. Development of up-to-date radiochemical technology in the world would increase the risk of nuclear weapon proliferation. Countries that use such a recycle, certainly, meet a problem of handling the radwastes produced in this process. To solve this problem as well as to utilize plutonium produced in thermal reactors and the excess weapon plutonium, released while reducing nuclear weapons, in the USA and Russia a closed fuel cycle concept is considered for thermal reactors and special burners are studied for burning plutonium, minor actinides and fission products, especially long-lived ones (iodine, technetium, etc). These activities are becoming now almost the only main directions of advanced reactor studies that can lead the nuclear power to a deadlock. In particular, in this way demands of the developing countries in significant increase of energy consumption cannot be satisfied.

If world population, as expected, doubles by the middle of the next century, it will cause a three-fold increase of the world demands in electricity production (curve 3 on Figure 1). Therefore, the role of the nuclear power in the first scenario based on thermal reactors remains very insignificant and countries that are extending their energy sector will have to solve their problems by developing other energy sources based, first of all, on organic fuel utilisation. However, this way can lead to a number of problems both specific for a country and global. The global problem related to organic fuel utilisation is ecological one. Insufficient resources of oil or gas in some developing countries can break a balance in the world fuel market and cause international political problems. The problem of fuel resources for large countries includes also the necessity to have a well-developed transport system.

Analysis of nuclear fuel balance shows that a thermal reactor of 1 GWe power capacity with ^{235}U fuel consumes 200 t of natural uranium per year. According to the first development scenario based on only thermal reactors, by the middle of the next century the thermal reactors will totally consume about 10^7 t of cheap uranium and generate about 10^4 t of fissile plutonium. This plutonium would allow to commission fast reactors of about 2 000 GWe with loading of about 5 t of fissile plutonium in a fuel cycle of a 1 GWe fast reactor with breeding ratio close to 1. This scenario of nuclear power development[1] corresponds to curve 2 on Figure 1. Availability of great amount of fissile plutonium eliminates the initial requirement to a fast reactor related to short Pu doubling time. Thus, breeding

ratio of the new fast reactors can be 1, no uranium blanket and high core power densities are necessary. Lead-based coolant can be used in these reactors instead of chemically active and low-boiling sodium coolant. Fuel types of high density and thermal conductivity can be used in these reactors. All these features can significantly enhance reactor safety and simplify design, i.e., decrease the reactor cost. Thus, the acceptable way to replace and stabilise organic fuel utilisation is to develop the large-scale nuclear power with new fast reactors and closed fuel cycle. These new fast reactors and closed fuel cycle should meet a number of requirements.

The new fast reactors, promising to be relatively cheap, should provide an acceptable safety level. The safety level is currently estimated by the probability safety analysis (PSA) that is, in fact, based on extrapolation of the available experience to future. Currently available experience of 10^4 reactor-years allows now to use the PSA methods for reactor safety analysis, in case, if the nuclear power develops by the first scenario based on thermal reactors. This scenario predicts increase in the experience up to about $5 \cdot 10^4$ reactor-years by the middle of the next century. However, if the large-scale nuclear power is planned to be created in the next century, the PSA methods cannot be sufficient for reactor safety assurance, because experience, i.e., input data for PSA, is insufficient.

Therefore, to provide reactor safety for the large-scale nuclear power it is necessary to develop a reactor concept based on the deterministic safety principles. One of such principals is a reactor inherent safety based on coolant and fuel properties, reactor self-regulation by feedbacks, coolant natural circulation for decay heat removal, etc. These reactor safety features should allow for deterministic elimination of dangerous development of severe accidents provided by any failures of equipment, errors of staff, or external effects, excluding, probably, a nuclear bombing attack. Such an approach estimates the reactor safety level as a maximum impact on the reactor that does not lead to unacceptable radioactivity release.

The requirements to the closed fuel cycle include the radioactivity balance, i.e., radioactivity extracted from the Earth in form of natural uranium should be equivalent to radioactivity returned in form of radwastes into underground disposals. To create the closed fuel cycle with radioactivity equivalent radwaste disposal, a non-aqueous fuel reprocessing technology should be developed with deep purification and with separation of fractions: fuel, including minor actinides, ^{129}I and ^{99}Tc, ^{90}Sr and ^{137}Cs. The neutron leakage from the fast reactor core could be used for transmutation of long-lived fission products (^{129}I and ^{99}Tc). ^{90}Sr and ^{137}Cs with half-life of about 30 years are supposed to be disposed in the storage until their activity is reduced by approximately a factor of 1000. The minor actinides are planned to be returned in the reactor with U and Pu after the reprocessing for burning. The main isotopes of Cm have short half-life, so, their removal from fuel for cooling could decrease fuel radioactivity. Low-active Np could also be extracted from the fuel as waste. The important feature of this closed fuel cycle, relating to non-proliferation problems, is that all its stages should not include plutonium separation. Thus, the possibility could be eliminated to use the power technologies for production of nuclear weapons. The weapon-grade plutonium is not generated in this fuel cycle because the new fast reactors do not have the uranium blanket.

Nuclear Power Unit with BREST-300 Reactor

Based on more than 30-year experience in operation of lead-bismuth cooled submarine reactors in Russia, a number of lead-cooled reactor projects of large, medium and small power capacity[2,3,4] was developed. A BREST-300 lead-cooled fast reactor[2] is one of the most developed medium-power capacity projects.

A BREST-300 lead-cooled fast reactor with U-Pu nitride fuel was developed in a number of Russian design and research institutions to obtain high parameters both for nuclear safety and economic efficiency. It is supposed that the BREST-300 reactor can be used as a heat source for generation of steam with high parameters, as a consumer of Pu obtained from reprocessing of spent fuel from thermal and fast reactors or weapons-grade Pu released in disarmament programs, as well as for burning of actinides and transmutation of long-lived fission products. Thus, this project can be a basis for solving problems formulated above. According to calculations, the reactor power of 300 MWe is minimum for providing core breeding ratio close to 1.

The reactor is a pool-type two-circuit steam-generating power unit (Figure 2) and includes core, eight once-through steam-generators of spiral-tube bundle type, four axial pumps, reloading system, control and safety system (CSS), turbine and emergency heat removal system. Thermal high-temperature protection and thermal insulation are used to minimise thermal leaks in normal operation. The supercritical water is the secondary working medium. The main technical parameters of the reactor is given in Table 1. Low pressure in the primary circuit and high parameters of the secondary coolant increase plant thermodynamic efficiency, simplify the reactor design, and increase operational reliability.

Low chemical activity of the coolant eliminates the danger of fire and explosion when primary coolant contacts the air or water. High density and thermal conductivity of the used U-Pu nitride fuel as well as filling of the fuel-clad gap with lead make it possible to significantly decrease the energy deposition and temperature gradients in fuel.

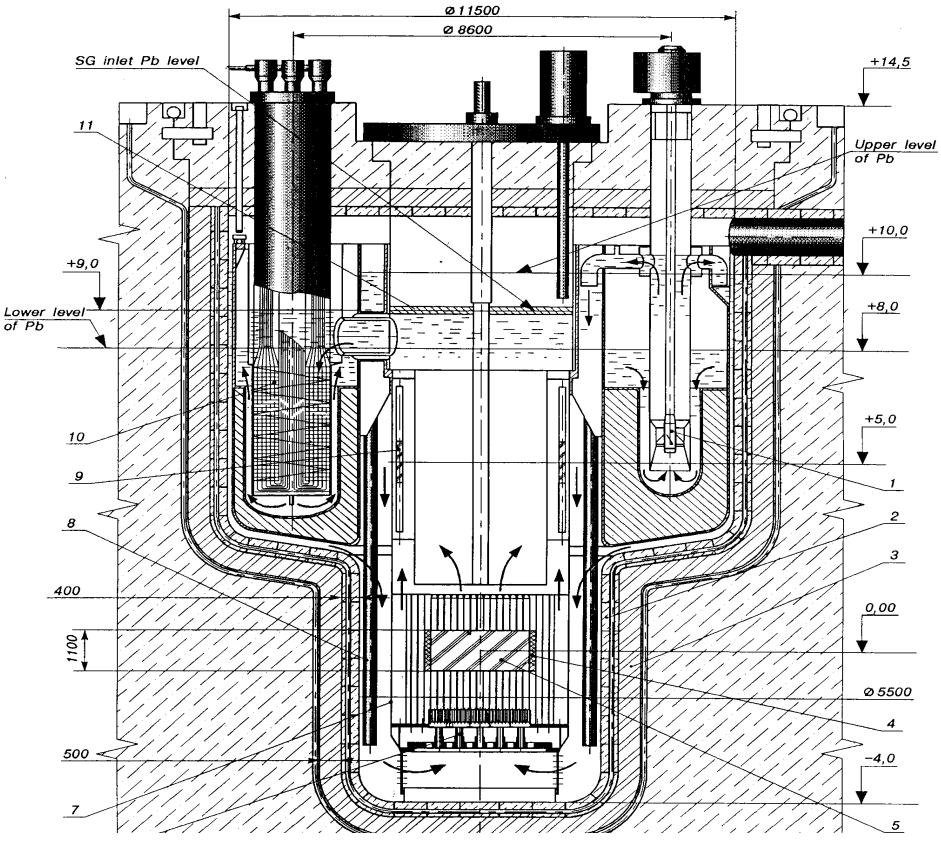
Coverless fuel assemblies are used to stabilise and flatten temperature distributions in the core. For the same reasons, simultaneous radial zoning of coolant flow and power density fields is used in the core. Each of three radial zones of the core (central, middle and outer) consists of fuel rods with different diameters with the same fuel composition and fuel rod pitch. Fuel rods with minimum diameter are used in central fuel assemblies and fuel rods with maximum diameter are used in outer zone. A rather large fuel rod pitch provides the large coolant flow cross section, small core hydraulic resistance, high level of power removed by lead natural circulation in emergency conditions (more than 10 %) as well as low coolant velocity (<2 m/s) and heating (120°C).

The lead circulates through the core and steam-generators due to the head created by the difference between “cold” and “hot” coolant levels created by pumps. Such a design eliminates flow irregularities in the core when one or several loops are disconnected. Moreover, it provides coolant circulation through the core during about 20 s after shutdown of all the pumps due to equalisation of the “cold” and “hot” coolant levels.

The supercritical parameters of the secondary coolant and feedwater heating up to 340°C prevent the lead coolant freezing (T_{melt}=327°C) in the steam generators when the reactor power decreases.

High boiling temperature of the coolant prevents the possibility of departure from nucleate boiling on fuel rods, overpressure shocks, as well as the possibility to lose the coolant. High specific heat of the lead circuit eliminates fast temperature increases in accidents. Even in a hypothetical accident with total loss of heat sink to the secondary circuit, increase in temperature of reactor components due to residual heat would not exceed 100°C in two hours after reactor shutdown. Nevertheless, an air heat removal system is provided to remove decay heat in emergency conditions. The system is based on air natural circulation in tubes installed in the downcomer and in thermal insulation concrete.

Figure 2. **BREST-300 reactor with pool-type configuration**



1. pump
2. high-temperature concrete
3. thermal insulation concrete
4. CSS
5. core
6. support pillars
7. dividing shell
8. air-cooled channels
9. spent FA storage
10. steam generator
11. rotating plug

A passive reactivity control system is provided for loss of flow accidents. When the lead flow drops to a specified level, absorbers enter the core. In normal operation these absorbers are supported in suspended conditions above the core by the hydrodynamic lead pressure head. Besides, a negative reactivity is inserted by decrease of lead levels in the CSS channels. A passive reactivity control system is provided also for accidents with increase of coolant temperature. Special thermal-mechanical bimetal spacers are located on the fuel assembly heads. When outlet coolant temperature increases, these spacers are thermally expanded and push apart the fuel assemblies. Thus, fuel assembly pitch is increased and additional negative reactivity is inserted.

The calculational studies of a set of transients[5,6] have shown that reactors possess high self-protection against the most severe accidents without scram: loss of coolant circulation in the primary and secondary circuits, lead overcooling at the core inlet, reactivity accident, failure of numerous steam generator pipes. The analysis of a hypothetical severe accident with failure of fuel rods showed that there is a basis to suppose that the secondary critical mass formation is eliminated due to the high lead density.

Table 1. **Main technical parameters of the BREST-300 reactor**

Parameter	Value
Thermal power, MWt	700
Electrical power (net), MWe	300
Primary coolant	Pb
Inlet/outlet lead temperature,°C	420/540
Steam generator outlet temperature,°C	520
Steam generator outlet pressure, MPa	24.5
Core layout	Square
Fuel composition	UN+PuN
Fuel loading (U+Pu)N, t	16
Radial and axial blanket	Lead reflector
Max. fuel burnup, %	Up to 12
Fuel assembly lifetime, years	~5
Reloading interval, years	~1
Peak/average fuel assembly power, MW	4.7/3.8
Peak fuel rod linear power, kW/m	44
Core diameter, mm	2300
Core height, mm	1100
Number of fuel assemblies in core	185
Lattice pitch, mm	150
Number of fuel rods in a fuel assembly	114
Fuel rod pitch, mm	13.6
Fuel rod diameter, mm	9.1/9.6/10.4

The main problems to be solved when developing the lead cooled reactor include study of corrosion of structural materials in lead flow, analysis of U-Pu nitride fuel properties, etc. A number of experimental studies, concerning heat transfer to lead flow, corrosion at high temperatures, critical experiments, etc. has already been conducted[7,8].

Conclusions

An advanced reactor concept was proposed to be a basis for development of the large-scale nuclear power in the next century. Inherent properties of the fast lead-cooled reactor make it possible to solve problems of nuclear safety, economic efficiency and radwaste handling, as well as to eliminate the possibility to use the power technologies for production of nuclear weapons. The solution was taken by MINATOM of Russian Federation to develop a technical project for construction of a pilot BREST-300 reactor.

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INNOVATIVE FUEL FORMS FOR BETTER MANAGEMENT OF NUCLEAR WASTE

T. Ogawa[1], J.S. Tulenko[2] and J. Porta[3]

1. Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken, 319-1106 Japan,

2. Nuclear Science Center, P.O.Box 118300, University of Florida, Gainesville, Florida 32611-8300, U.S.A

3. CEA Centre d'Études Nucléaires de Cadarache, DRN/DER/SIS 13108 Saint-Paul-lez-Durance, CEDEX, France

Abstract

Integration of designing the front end (manufacturing/operation) and that of the back end (waste management and/or recycle) is becoming an important issue in the nuclear technology. Innovative fuel forms have been proposed and studied in view of better waste management and resource conservation. The fuel forms, which satisfy those newer requirements without penalising the economy of the fuel cycle, have yet to be established. Some of the concepts, which are currently investigated, are introduced, technical issues are identified, and common material problems are discussed.

1. Introduction

There are accumulations of plutonium and minor actinides (MA: neptunium, americium and curium) in spent fuels worldwide. Plutonium may be recycled as mixed-oxide (MOX) fuels to the power reactors. However, it is not the ultimate solution to the problem of plutonium management, since the spent MOX fuel still contains a significant amount of plutonium and MA. Despite the worldwide use of oxide fuels in commercial reactors, their limitations as well as advantages have been identified. In parallel with R&D for improvement of the oxide fuel performance, the search for innovative fuel forms would be required with further emphasis on the management of actinides in view of environmental and political concerns. Although the technology alone cannot solve the problem, it may, at least, adopt itself to the changing socio-political conditions.

The industry should engage in life-cycle design. Life cycle design means that the selection of a fuel type requires an evaluation of the back end of the fuel cycle (including non-proliferation items) equally with the front (manufacturing) end and the operational (fuel performance) phase. The back-end has often only been considered after the fuel type has been selected, but not as a part of the selection process. The neglecting of the back-end when choosing a product design is not unique to the nuclear fuel, but, unfortunately, a common flaw in the design process.

2. Innovative fuels trends

The two overriding considerations in fuel design and performance are cost and safety. These have been well studied in the performance end of the current power-reactor fuels, but have not been addressed for the back-end, i.e., disposal and recycle.

Depending on each nation's nuclear policy, there are obviously two options for the innovation in nuclear fuel technology:

- the fuels for improved spent-fuel waste management,
- the fuels for improved actinide burning or actinide recycling.

In the first category, we have the technological targets:

- ultrahigh-burnup to reduce the number of spent fuels,
- better immobilisation of radioactivity in geologic disposal.

In the second category:

- more efficient actinide burning and/or breeding,
- improved economy and proliferation resistance of fuel cycle.

Although the intent and technological targets appear to be diverse, technical issues are connected to each other. For instance, both the ultra-high burnup and the efficient actinide burning/breeding would require fuels with reduced fission-gas release and swelling. Conventional power reactor fuel is characterised by a steep temperature gradient: the oxide fuel pellet is cold outside ($T/Tm < 0.3$), but hot inside ($T/Tm \sim 0.5$). Evolution and interplay of diverse processes at different temperature regimes make it extremely difficult to extrapolate the fuel performance data to the ranges beyond the past experience. There may be potentials for ultra-high burnups by simplifying this situation by using either "Cold" or "Soft (very hot or molten)" fuel. The cold fuel reduces fission gas release and swelling; the hot fuel like metallic fuels in fast reactors allows fission gas release and thereby eliminates the mechanical interaction by swelling. We may also expect additional benefit of improved safety (LOCA for LWR and ULOF for FBR etc.).

Another keyword is “Integration”: better economy and waste management could be realised by integration of the fuel forms with either recycling (IFR-type approach) or disposal (fuels mimicking natural minerals). The chemical state of the burnup fuel would be tailored to each fuel-cycle scenario.

However, we have to admit that the technical solution depends on the available neutron field, which is a function of the nuclear policy and the economical condition (uranium price, resource availability, energy-security cost etc.). Therefore, it would be proper to divide the further discussions into two parts:

- fuels for thermal reactors,
- fuels for fast-neutron field.

3. Innovative fuels for thermal reactors

Innovative fuels for thermal reactors are being studied to improve the spent-fuel waste management and resource conservation. The approach led to the search for the ultrahigh-burnup fuels. In countries where the nuclear policy requires the recycled use of Pu, the better way to suppress the separated Pu inventory would have to be sought in view of the delay of the FBR programs. Regardless of the fuel-cycle policies, those approaches converge to the quest for innovative nuclear materials such as those for cladding and metallic thermal bonding (or CERMET fuel). The life-long reactivity control is another important subject. A few examples of the efforts in this direction are given below.

3.1 Ultra-high burnup

The development of ultra high burnup in light water reactors is driven by either national nuclear policy or by economics. Its development is challenged by the level of technical performance reliability required by light water reactor fuel that mandates improved cladding corrosion, better fission gas management, and better lifetime performance of all materials in general. In the United States ultra high burnup studies are driven by the once-through cycle that requires that fuel will be once burnt and then permanently disposed. This requirement places a strong incentive on obtaining the maximum efficiency from the fuel while minimising the amount of fuel to be disposed.

Unlike a fast reactor which can breed new fuel and only needs to offset the reactivity decrease due to fission products, uranium fuel for light water reactors has a declining reactivity curve that has forced fuel designers to focus on poison management to achieve high burnup while staying within the operational constrains of the reactor control system. A successful model of what can be achieved is available. Ultra high burnup fuel is being designed for the US naval light water reactor program that will enable the core to operate for the life of the plant so that the ship does not need to refuel. As recently reported by Guida et al.[1], nuclear powered warships currently operate more than twenty years without refueling. An example is given of the USS NIMITZ that went to sea in 1975 and will receive its first and only refueling in 1998. The paper reports that for the next generation of submarines the navy is developing nuclear fuel that will last as long as the life of the ship, about 30 years. It is noted that the navy fuel fully contains all fission products after 20 to 30 years of operation. It was pointed out that such fuel is extremely well suited for safe transport, storage, and ultimate disposal. Such a capability greatly reduces the proliferation concerns since the fuel is either incore or being permanently disposed. Also, as pointed out earlier, the navy fuel is exceptionally good in providing a first level engineered containment system for fission products. However, the naval nuclear program is based on military requirements and has not been based on economics. The navy is currently moving away from nuclear powered warships for economic reasons along with a belief that long life is no longer a military need. However, the navy program has demonstrated the technical feasibility of long life fuel and given a target for all to strive for.

In the United States ultrahigh burnup high efficiency fuel is being proposed to reduce the volume of spent fuel that will have to be disposed.[2] Currently, concerns on the length of time required to store ultrahigh burnup fuel to reduced stored heat prior to disposal and the need to demonstrate the reliability of the performance of ultra-high burnup fuel particularly under accident conditions have raised challenges to the creation of a national program to investigate fuel burnups above 70 GWd/tHM. The proposed United States Programs will explore a variety of new or improved materials to replace the current generation materials for higher or more efficient burnup.[3] In addition to the development of new alloys of zircaloy, totally new cladding materials such as composites of silicon carbide or other ceramic composite are being proposed. The key to ultra high burnup however, will be the development of an economical long-lived burnable poison system. In light-water reactors the oxide fuel form tends to be mandated by the solubility of non-oxide forms in water. The success to date of UO_2 in light-water reactors has discouraged the investigation of other forms of thermal reactor fuel. The grain structure of UO_2 is being researched in hopes of finding a structure which will yield better irradiation performance and fission gas retention. However, as mentioned earlier the relatively poor thermal conductivity of UO_2 fuel has prompted research to find designs or materials to reduce the centreline temperature. Annular fuel pellets, liquid metal bonds and metal fuel are among concepts that have been investigated to lower the operating fuel temperature.[4,5] In other countries, higher burnup is being proposed to offset the cost of recycling plutonium and the ability of plutonium recycle fuel to achieve a flatter reactivity curve. In any country, the development of on-line maintenance capabilities which allows the continuous operation of the plant will be a major incentive towards ultrahigh burnup fuel.

The particular problems associated with higher burnup, that the replacement materials are being sought for, are: cladding corrosion, fission gas release, fuel swelling, cladding ductility, structural strength and radiation growth. Bonding and coating materials are being looked at to see if they can contribute to acceptable performance at higher burnups. New fuel materials are being reviewed which will increase the fuel loading while also increasing the fuel thermal conductivity and allowing fission gas retention. Fast reactors and the naval reactors program have shown that ultrahigh burnup can be achieved. However, the materials that allow ultrahigh burnup today in these plants are not economic in commercial light water reactors. Therefore, the search goes on for materials (fuel and cladding) that will operate safely and economically to ultrahigh burnups and allow for economical and efficient reprocessing or disposal.

3.2 Inert Matrix Fuel

In France, due to a delay in the FBR program, it would be necessary to manage a new reality with the Pu inventory. The present and future 30 per cent MOX loaded PWRs are not able to decrease the French Pu inventory. We would have to consider new solutions to increase the Pu loading and burning in PWRs. This may be achieved by using a uranium-free fuel to avoid the conversion of ^{238}U into ^{239}Pu. Two types of composite fuels are studied: CERCER and CERMET. The (oxide) Pu fuel is a ceramic (CER), which is dispersed with an adequate volumetric ratio in an inert matrix which is either a ceramic (CER) or a metal (MET).

Two aspects have been studied: the theoretical one with optimisations of the neutronics, and understanding of the fission gas release mechanisms[6-8] and the technological one with the study of the materials behaviour under irradiation. The first irradiation test has been made in the TANOX device in the SILOE reactor (Grenoble). The CERCER sample showed a significant swelling of the spinel ($MgAl_2O_4$) matrix, while the CERMET showed a good dimensional stability. A confirmation and an explanation of the spinel swelling in PWRs conditions have been given by Matzke.[9] Since then, studies have been pointed toward CERMET.

Owing to a very good thermal conductivity, the CERMET allows a very low centre pin temperatured with a moderate gradient inside. In the TANOX conditions, the centre pin temperature was 540°C for a linear power of 350 W/cm, that leads to something close to 450°C in PWR is conditions. Obviously, it is an important parameter to limit mechanical stress and fission gas release.

After having analysed these results, a second experiment has been made with a CERMET fuel including a low content Er_2O_3 in the ceramic fuel component, in order to simulate a U free Pu fuel, which must be controlled by a burnable absorber to lower the initial reactivity. The fuel reached a burnup of 130 GWd/t. At present, measurements of fission gas release have yet to be made, but the excellent dimensional stability of the pellets has been verified.[10,11]

Important neutronic and thermohydraulic studies have been performed in order to define sub assemblies and cores allowing a maximum Pu loading. The major problems are connected to the control of the potential reactivity. We have to deal with the very low effective Beta and Doppler coefficient, and the real possibility to obtain positive values for the voiding effect and the moderator coefficient at the beginning of cycle. The U-free Pu in a 100 per cent inert-matrix fuel core loading represents an academic case to study the fundamental physics of the fuel, but this solution does not seem to be realistic.[12,13] Alternative solutions have been proposed: some very interesting solutions have been presented by the scientists of ENEA (Italy) and PSI (Switzerland) in the framework of the Inert Matrix Fuel Workshop.[14-16]

The CEA initiative in this field has converged toward a bi-heterogeneous sub-assembly concept: the first heterogeneity is a spatial one, the second concerns the fuel. In this sub-assembly there are the standard UO_2 and the CERCER pins (at present PuO_2 in a CeO_2 matrix). The CERCER fuel pins are annular, large in diameter (~26 mm) and very thin (thickness ~1.3 mm), with an internal and external Zr_4 alloy cladding to have an additional inner water passage in order to improve the cooling. In addition, the local moderation ratio is increased; the kinetic parameters, the boron worth and the Pu utilisation are improved.

This APA (Advanced Pu sub-Assembly) concept leads to minor modifications in the core and in the control system of a standard PWR. It has been shown that with less than 30 per cent of APA-loaded PWRs it will be possible to stabilise the Pu inventory in the French nuclear plants.[17-20] The R&D program focuses now on the optimisation of the manufacturing process of the APA pellets, the definition of the transients in the mechanical and thermomechanical behaviour of the core, and the APA fuel (and core) behaviour under severe accident. In fact this concept is very versatile and APA is able to accept any kind of fuels with flexible spatial dimensions in order to adjust the right moderation ratio to the fuel.

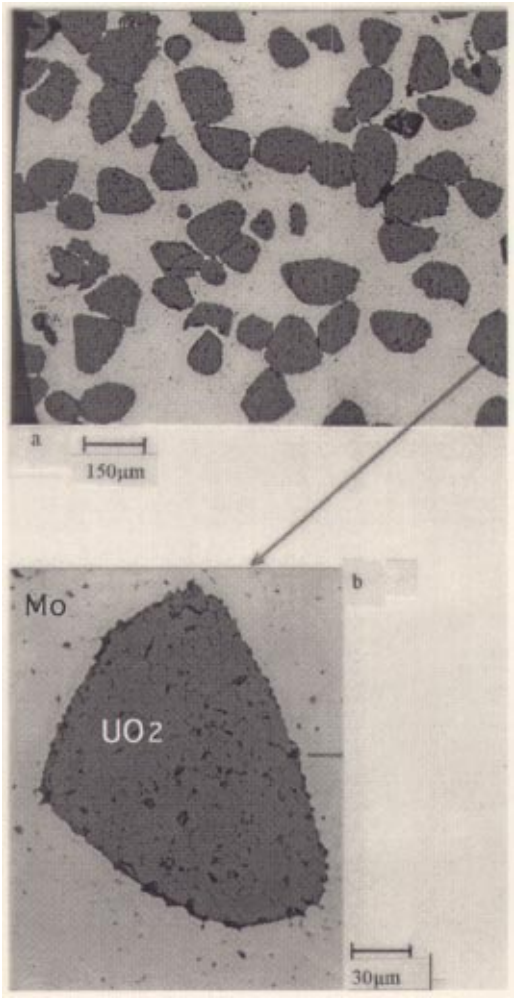
The second part of the program is devoted to the CERMET fuels. Preliminary studies[21,22] have shown that owing to the thermal conductivity and the metal-metal contact between the cladding and the matrix, in case of severe accident (loss of coolant), it is difficult to reach the melting temperature of the cladding and the fuel. Similar results have been obtained in Russian on CERMET fuels in VVER reactors.[23] It may be possible to keep a cladding temperature under the departure of the very exothermic kinetics of Zr_4 oxidation.

The first goal of the CERMET fuel program was to increase the Pu loading in a PWR core. The results of the experiment have also shown a way to achieve ultrahigh burnup discussed in 3.1. In fact the CERMET fuel allows a very high burnup (up to 130 GWd/t) without important damages to the matrix and with an excellent fission gas retention. The metal matrix adds an extra barrier in the safety concept. It will be an excellent way to obtain an once-through cycle, but it is also important to note that this fuel may be readily reprocessed. One may separate the fuel and the metal matrix (50 to 80 per

cent of the total volume), which can be recycled for new fuel fabrication. In the case of a Zr_4-PuO_2 CERMET, in a PWR, starting at BOL with a Pu vector of (56 per cent ^{239}Pu-7.4 per cent ^{241}Pu), after a 12-month cycle we obtain at EOL a Pu vector of (4.5 per cent ^{239}Pu-18 per cent ^{241}Pu) to compare with a MOX fuel having a EOL Pu vector of (43.5 per cent ^{239}Pu-14.3 per cent ^{241}Pu). This CERMET fuel allows to achieve three goals:

- to improve the fuel cycle costs by realising a very high burnup;
- to burn Pu fuel as well as minor actinides, with good safety margins;
- to reduce the volume of waste significantly.

Figure 1. **CERMET (Mo-UO_2) irradiated and heated at 1853K for 30 minutes**



3.3 Rock-like fuel (ROX)

As already discussed in 3.2, better management of plutonium may be realised by burning plutonium in the once-through fuel cycle with a uranium-free fuel. Further improvement in view of spent-fuel management is expected by using a fuel matrix where fission products and actinides are chemically fixed. The concept of using rock-like oxide fuel (ROX) in light-water reactors was brought out by the Japan Atomic Energy Research Institute (JAERI) under these reasonings.[24,25] The approach has a parallel in designing highly durable nuclear waste by the use of natural phases of great age as

proposed by Ewing.[26,27] The latter author proposes the materials designing for waste disposal with more emphasis in "immobilisation" or "containment" in addition to the performance assessment on "geologic isolation".

The name ROX comes from the basic idea of mimicking the natural phases to assure the long-term prediction of the spent fuel behaviour in geologic conditions. Several combinations of ceramic phases have been tested. At present, the candidate fuel is designated Zr-ROX, whose matrix is composed of stabilised zirconia [ZrO_2 (Y, Gd, (Er)] and spinel ($MgAl_2O_4$). The zirconia will dissolve actinides and rare earths; the spinel will fix fission products of alkali and alkaline-earth metals such as Cs and Sr. Erbia may be added to the zirconia matrix to reduce the burnup reactivity swing.

However, a completely uranium-free fuel in LWR would not be viable. The negative Doppler coefficient of uranium-free core is too small, and the addition of a resonant nuclide, ^{232}Th or ^{238}U, is necessary. An alternative is to load the Zr-ROX fuel partially in the UO_2 core. Table 1 compares the various schemes of loading plutonium in PWR.

Table 1. **Net transmuted percentage of initial plutonium inventory for various loading types in PWR. (Core thermal output: 3 411 MWth × 1 170 EFPD)**

		weapon-Pu Transmuted %	reactor-Pu Transmuted %
1/3ROX+2/3UO_2	^{239}Pu	75	67
	total Pu	43	41
(in 1/3ROX only)	^{239}Pu	99	98
	total Pu	86	74
Zr-ROX(Er)-UO_2	^{239}Pu	92	88
	total Pu	69	60
Zr-ROX(Er)-ThO_2	^{239}Pu	97	93
	total Pu	79	66
Full MOX	^{239}Pu	63	46
	total Pu	30	25

Further optimisation of the fuel matrix and the core compositions are due in the future R&D. The stabilised zirconia will be, however, regarded as the major component of ROX. The issues of irradiation stability of inert matrices are discussed in paragraph 6.

4. Innovative fuels for fast-neutron field

In view of the fuel economics in the early part of the 21st century, the first role of the fast neutron field might be burning rather than breeding actinides. There are various concepts for burning Pu and/or minor actinides (MA) in a fast neutron field. We would discuss two examples here.

4.1 Pu and MA burning in fast reactors

In France, the CAPRA program[28] is devoted to the study of the Pu and long-lived radioactive wastes [minor actinides (MA) and fission products] burning in fast reactors. This program is a part of the most generic one where the burning of Pu, MA and other fission products in conventional reactors or innovative devices optimised for this purpose are widely studied.

The reprocessing and the utilisation of Pu are scheduled in order to ensure continuity between the present nuclear plants, PWR in France, and the future ones with the most important part composed by fast reactors. This continuity may be ensured by the progressive introduction of CAPRA reactors on the grid. Their versatility in terms of breeding gain allows to look forward the regulation of the Pu inventory during the transient step between a 100 per cent PWR grid to a mixed PWR-FBR one.

In this context three basic scenarios are defined:

- a whole FBR grid, the self breeder reactors manage themselves their Pu, MA and fission products;
- a mixed grid composed by a part of FBR. The 4/94 CAPRA reactors manage Pu, MA and fission products of all the reactors of the grid;
- one of the part of the grid is constituted by innovative systems optimised to burn long-lived MA and fission products. The Pu management is performed in the PWR (PWR with MOX with a slightly enriched U or using APA concept.), and/or in the FBRís. The MA and FP are burnt in a burner fast reactor dedicated to the MA burning ("double-strata" according to the Japanese terminology).

Scenarios 1 and 2 lead to investigate the burning of MA and LLFP in the CAPRA 4/94 core (which can be seen either as a burner or a breeder reactor), which had previously been optimised for high Pu consumption. All the transuranium elements and LLFP are managed in the same type of reactor.

MA can be burnt by homogeneous recycling (i.e. with the MA diluted in the CAPRA fuel). In this case, high Pu content oxide fuel, enriched with MA up to 10 wt per cent, is of a special concern in the French context of PUREX reprocessing. The metallic fuel and its specific cycle involving pyrometallurgical processes also become very attractive and deserve to be studied.

The mixed oxide fuel with a high Pu content, which is the reference concept for burner Pu core, has been tested in-pile. Some encouraging results are now available.[29,30] The consequences of the introduction of the small MA content (< 2.5 wt per cent) in the pelletized MOX fuel have been assessed.[31] For a larger fraction of MA, a program on VIPAC mixed oxide fuel is investigated in the BOR 60 reactor within the MINATOM co-operation.

The greater part of the French program is focused today on the heterogeneous recycling (i.e. with the MA concentrated in specific targets, separated from the CAPRA fuel)[32,33] MA-based targets are optimised to be mono-recycled (once-through) or multi-recycled. In both cases moderated targets are used to enhance the MA consumption without reaching a too high damage dose on cladding materials.[34] A progressive approach has been chosen to prove the feasibility of the heterogeneous recycling. The first step is to optimise oxide CERCER targets (mainly $MgO+AmO_{2-x}$) as a safer option because of the French know-how in oxide fuel cycle. The irradiation test in the Phenix reactor has been launched for improving the reference $MgO+AmO_{2-x}$ concept.[35,36] The next steps are to assess more innovative concepts (like nitride or metal targets).

The third scenario leads us to conceive very innovative core and fuel. The MA-based fuel with a MA content as high as possible (typically 50 to 70 wt per cent of MA, the balance of heavy atoms being Pu or enriched U) is completely new. Compared with the oxide CAPRA fuel, the MA compounds based on the oxides are expected to have poorer thermal properties (stability, conductivity, melting point, etc.). Also the helium release, due to decay of ^{242}Cm produced by transmutation of ^{241}Am, is expected to be very high.

Because basic data on MA compounds are missing, it is not possible today to be sure that the MA-based fuels are feasible. In the frame of the CAPRA project, involving a large European co-operation, the PIMPOM irradiation experiment scheduled in HFR, Petten, in 2001 is the first step to test the feasibility of the MA-based fuels/targets. Different nitride fuels (solid solutions like (Pu, Zr) N, CERCERs like PuN+TiN, CERMETs like PuN+W, etc.) will be irradiated to a high burnup in a short time (typically 20 at per cent burnup in one year). Because americium is not available today in sufficient quantity, Pu is used as a surrogate.

4.2 Nitride fuel and double-strata fuel cycle concept

The actinide mononitrides are characterised by a high heavy metal density, a large thermal conductivity and a high melting point. Good thermal properties give a "cold" fuel, if we adopt a metal-bonded (normally Na-bonded) fuel element, where both fission-product gas release and swelling are expected to be sufficiently low to a very high burnup. The "cold" fuel also gives a negative Doppler term during accidents such as ULOF. The major drawback of the nitride fuels is the use of highly ^{15}N-enriched nitrogen, which would have to be used to minimise the formation of ^{14}C by the (n, p) reaction of ^{14}N.

The mononitrides of actinides have the NaCl-type cubic structure. Their lattice parameters are close to each other, and we may have homogeneous mixtures of actinide mononitrides for various combinations and compositions. This is a unique advantage of using nitrides as the fuels for the actinide burning. However, the properties of AmN and CmN are scarcely known. Further study on these compounds is required. On the other hand, there is a database on the thermodynamic and thermophysical properties of UN, NpN and PuN.[37-40]

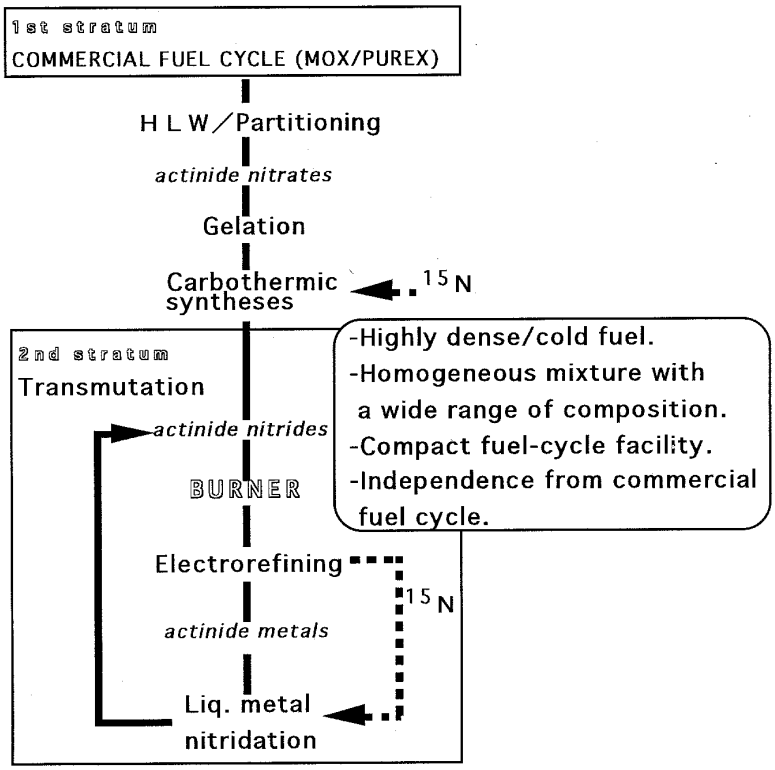
The irradiation behaviour of (U, Pu) N fuel elements has been studied in detail[5,6], though the database is limited to burnups below 10 at per cent. In an irradiation test in EBR-II, Na-bonded (U, Pu) N test fuel pins achieved over 9 at per cent burnup without failures.[41] Besides, if we can transfer the experience on the Na-bonded carbide fuels to the Na-bonded nitride fuels, FCMI can be avoided to burnups over 11 at per cent.[38] Actually the nitrides proved to be dimensionally more stable than the carbides. Still the difference in the mechanical properties between the carbides and the nitrides may have unpredictable effects at high burnups.

In JAERI, the actinide burning based on a double-strata concept is being studied, which consists of the first stratum "commercial fuel cycle for U and Pu" and the second stratum "MA-burner fuel cycle".[42] The MA burner may be either a dedicated reactor or an accelerator-driven system with a sub-critical core of the MA nitride (MAN) fuels. The double-strata approach to the transmutation of MA and long-lived fission products has advantages:

- the practices in the commercial fuel cycle is little affected by the addition of the second stratum;
- the transmutation can be made regardless of the schedule of fast-reactor introduction;
- the fast reactor technology can be further developed within the framework of actinide burning.

The MAN fuels may be reprocessed by a pyroprocess[43] similar to that proposed for the U-Pu-Zr metal fuel by the Argonne National Laboratory, U.S.[44] Although the applicability of molten-salt electrorefining to the nitrides has yet to be explored in detail, the initial attempts of recovering Pu and Np from PuN and NpN, respectively, were successful.[45,46] The advantage of applying the pyroprocess is the ease in recycling ^{15}N: the dilution of ^{15}N by natural nitrogen can be avoided by treating the spent nitrides in molten-chlorides media.

Figure 2. **Process scheme of MA recycling and double-strata fuel cycle concept**



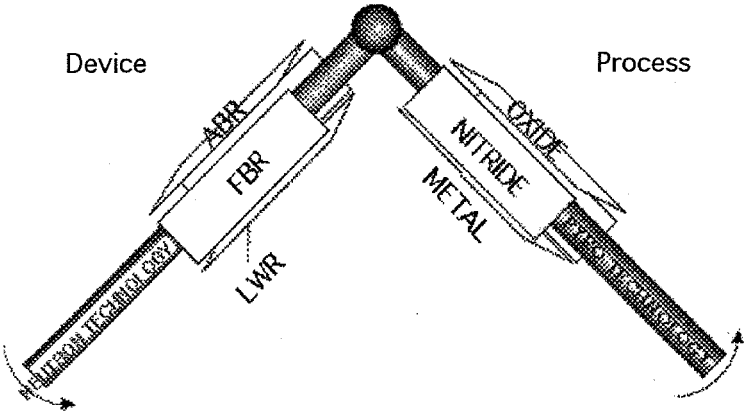
5. Integration and coupling by a common technology

Since the IFR concept has been proposed by ANL, "Integration" is often a keyword in discussing the future fuel cycle. The pyroprocess, such as the electrorefining of metal fuels, is widely regarded as a viable future option to minimise the fuel cycle cost, even though most process components have yet to be demonstrated in engineering scale. There has been a severe criticism: a review[14] pointed out the difficulty of materials accounting in the IFR process, which implies a significant limitation in the safeguarding practice. It may not be easy to answer such a criticism at this stage. However, the characteristics of the pyroprocess of being economically competitive in a smaller production scale, may make an evolutionary approach feasible. We may have modular fuel-cycle facilities of smaller scales, which will be optimised and standardised with time and experience for better economy and safety. Then, it would be easier to deal with changing social and economical situations surrounding the nuclear technology. The module would eventually evolve into the standardised one, which makes both the safety and safeguarding practices simpler.

"Integration" should not imply a rigid combination of key components. Rather it should be regarded as a flexible coupling of components by a common technology to make a whole. The fundamental process should be versatile enough to be compatible with different types of fuels (nitrides, oxides and alloys with or without minor actinides). The IFR-type pyrochemical process may be regarded as an example of such a common technology.

We can list such common technologies which would become bases for the better future of nuclear technology. The other potential technology is the inert-matrix technology (CERCER and ROX), which has many features in common with the new waste form development. The cold fuel concepts, as represented by metal-bonded or CERMET fuels as well as nitride fuels, warrant further studies also.

Figure 3. **Integration/coupling through common technologies**



6. Difficulties and future direction

As far as the fuel performance is concerned, the following fundamental issues have to be further addressed:

- Effect of the existence of MA in higher concentrations than hitherto experienced. The behaviour of decay-product helium for instance has yet to be clarified. In both the burnup extension of MOX and the use of Am-bearing fuels, helium production from transmutation products such as ^{242}Cm could reach a very high level.[48]
- Effect of fission-fragment damage on ceramic matrices at very high doses. Already, among the candidate inert matrices, alumina (Al_2O_3), zircon ($ZrSiO_4$) and monazite ($CePO_4$) have been discarded due partly to the instability against fission-fragment damage.[49] Also the science of "cold fuel", whose ultrahigh-burnup behaviour may not be necessarily predicted by a simple extrapolation from low burnups, has yet to be explored.
- Physical and chemical database of transuranium elements and compounds. Those on actinide oxides have been most extensively studied. The problem of a high oxygen potential over AmO_{2-x} has been identified[50,51], although its implication in the Am-bearing fuel performance has yet to be clarified. The high vapour pressure of Am would be a problem in the fabrication of Am-bearing nitride or metallic fuels.[37]

Bringing out an innovation in the long-term actinide management might not be an urgent issue and might be even considered to hinder the solution of more urgent issues. At the same time, it is not very clear if there is a real way out of the actinide problem without introducing any innovative measures. The current safety logic being developed with the history of nuclear technology is rigorous, and it does not necessarily take into account a completely new technological approach. This factor alone would be enough to make the local management to avoid the innovation. The formation of a kind of "hutch" would be needed for innovative technologies, but it has become a very difficult task for any single organisation. While the fuel and fuel-treatment technologies are pivotal in developing the future nuclear systems, R&D on them are costly and time-consuming. It would be more advantageous to allocate the R&D resources in an internationally concerted manner. The technological "hutch" would not have to be concentrated in a single location, but it may be an entity on the network connecting the distributed R&D resources.

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A CLOSED THUOX FUEL CYCLE FOR LWRS WITH ADTT (ATW) BACKEND FOR THE 21ST CENTURY

Denis E. Beller, William C. Sailor, and Francesco Venneri
Los Alamos National Laboratory, Los Alamos, New Mexico, U.S.A. 87545

Abstract

A future nuclear energy scenario with a closed, thorium-uranium-oxide (ThUOX) fuel cycle and new light water reactors (TULWRs) supported by Accelerator Transmutation of Waste (ATW) systems could provide several improvements beyond today's once-through, UO_2-fueled nuclear technology. A deployment scenario with TULWRs plus ATWs to burn the actinides produced by these LWRs and to close the back-end of the ThUOX fuel cycle was modeled to satisfy a U.S. demand that increases linearly from 80 GWe in 2020 to 200 GWe by 2100. During the first 20 years of the scenario (2000-2020), nuclear energy production in the U.S. declines from today's 100 GWe to about 80 GWe, in accordance with forecasts of the U.S. DOE's Energy Information Administration. No new nuclear systems are added during this declining nuclear energy period, and all existing LWRs are shut down by 2045. Beginning in 2020, ATWs that transmute the actinides from existing LWRs are deployed, along with TULWRs and additional ATWs with a support ratio of 1 ATW to 7 TULWRs to meet the energy demand scenario. A final mix of 174 GWe from TULWRs and 26 GWe from ATWs provides the 200 GWe demand in 2100. Compared to a once-through LWR scenario that meets the same energy demand, the TULWR/ATW concept could result in the following improvements:

- depletion of natural uranium resources would be reduced by 50 percent;
- inventories of Pu which may result in weapons proliferation will be reduced in quantity by more than 98 percent and in quality because of higher neutron emissions and 50 times the alpha-decay heating of weapons-grade plutonium;
- actinides (and possibly fission products) for final disposal in nuclear waste would be substantially reduced; and
- the cost of fuel and the fuel cycle may be 20-30% less than the once-through UO_2 fuel cycle.

Introduction

As the world transitions from a rapid growth in nuclear power during the last half of the 20th Century to a changing 21st Century, we search for a nuclear fuel cycle with sustainable, economic, and environmentally sound characteristics. A fuel cycle that includes abundant thorium has been studied extensively in the past [1] but has not been deployed nationally for several reasons. These reasons include, but are not limited to, the following:

- Uranium has been a relatively inexpensive and an abundant fuel for safe nuclear power worldwide.
- In the U.S., at least, a "once-through" conventional LWR (CLWR) fuel cycle has been employed as most economic.
- Used fuel from the CLWRs has been stored on-site in "spent-fuel" pools, and on-site storage space has not yet been exceeded.
- There are potential limitations associated with geologic disposal of waste from the once-through U-based fuel cycle that have not been demonstrated or discovered because lessons cannot be learned from experience until a repository is constructed and waste is being packaged and stored.

Although the uranium resource is abundant, it is not inexhaustible. Non-nuclear energy resources are also not unlimited, and a worldwide realization of the hidden costs of other energy sources or a global consensus to reduce greatly the emission of greenhouse gases and other pollutants could result in a resumption of rapid growth of nuclear power similar to the 1970s in the U.S. Thus, the world may once again face the issue of limitations on economical uranium resources, as well as the continuing and escalating issues of waste disposal. Another issue is beginning to take on additional meaning, especially with recent global activities in development and deployment of weapons of mass destruction. That issue is proliferation of nuclear weapons capabilities, which requires "weapons-useable" materials for use in manufacturing those weapons.

Continued worldwide studies of these issues and how to solve them have revealed that an advanced fuel cycle may have a great potential to reduce resource depletion, to reduce quantities and improve the composition of waste for disposal, and to reduce the risk of proliferation of nuclear weapons. To examine the potential impact of a national deployment of an advanced fuel cycle, we studied a system similar to one that is currently receiving heightened attention worldwide: a combined thorium-uranium-oxide (ThUOX) fuel for new LWRs (TULWRs). The Department of Energy has recently funded a joint U.S./Russia project to examine a "seed/blanket" fueling option – called Radkowsky Thorium Fuel (RTF)[2] – for existing LWRs that employs a mixed Th-U fuel in the blanket assemblies. However, the Th-U fuel in the present systems analysis was not restricted to the RTF fuel cycle, but was modeled with a "generic" ThUOX cycle. In addition, to minimize the impact on the environment, and to reduce quantities of waste and weapons materials, the front end of the ThUOX fuel cycle was combined with a back-end accelerator-driven transmutation technology (ADTT): LANL's conceptual Accelerator Transmutation of Waste (ATW) system.[3] The ATW will be used to improve the fuel cycle by reducing quantities of actinide and possibly fission-product waste for disposal and by eliminating most of the Pu inventory in the U.S. to reduce the risk of proliferation. This coupled concept of TULWRs and ATWs produces a view of the future with greatly reduced natural resource depletion and waste volume, and with a reduction in proliferation risk, in terms of both quality and quantity of plutonium stored external to the highly radioactive and controlled cores of reactors.

The TULWR/ATW fuel cycle uses less natural-U resource than the once-through CLWR because the reactors will use thorium for 70-90% of the fuel, and they recycle ^{233}U (bred from Th), so they use a smaller quantity of enriched uranium and much reduced natural uranium. The system produces less Pu because the fuel contains less U, and the Pu is of lower "quality" for proliferation because recycling of intermediate isotopes of uranium produces more ^{238}Pu and other "non-fissile" isotopes of Pu. The combined system also generates less radioactive waste for disposal because long-lived fission products (LLFP) and actinides are separated from spent fuel and transmuted to short-lived fission products in ATWs. An additional benefit is that this processing produces a stream of separated isotopes which may be in optimal chemical forms for disposal.

In the following sections of this paper, the previous statements about this combined, ThUOX-fueled TULWR and ATW system for the production of nuclear-generated electricity out to the year 2100 will be quantified, and non-proliferation attributes will be explored. We begin by describing the scenario that defines the growth in nuclear energy and the transition to TULWRs supported by ATW backend.

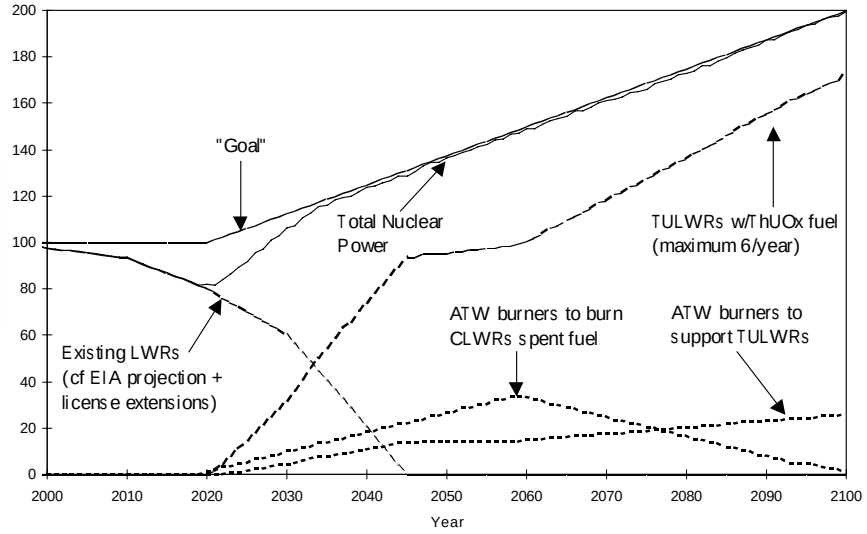
Basis Scenario for Transition – Today to 2100

In 1997, The Energy Information Administration of the U.S. DOE examined the outlook for nuclear power in the U.S. out to the year 2020, with a "High nuclear", a "Reference case," and a "Low nuclear" scenario.[4] The high nuclear case is characterized by extension of licenses for all existing plants by 10 years, and the reference case by some shutdowns and some reactors retiring at the end of existing licenses. The low case assumptions include the 10-year-early retirement of all existing nuclear power plants in the U.S. In view of ongoing de-regulation of electricity generation, distribution, and sales; recent applications for 20-year license extensions; and industry attempts to purchase U.S. nuclear power plants; for this study the "high nuclear" projection of the EIA was used, although this scenario may be approximated by some early retirements combined with 20-year license extensions and improved performance and efficiency. Thus, as is shown in, during the first 20 years (2000-2020) of the scenario included in this study, nuclear energy production in the U.S. declines from today's 100 GWe to about 80 GWe in 2020. For the period beyond the EIA study (after 2020), all existing reactors are shut down by 2045. Also, for the present analysis neither Evolutionary LWRs nor any other nuclear systems to supply either additional needed power or to make up for retired nuclear generation during the next 20 years would be deployed (realization of this assumption, which is in accordance with the EIA assessment, would have a substantial negative impact on efforts to reduce the emission of greenhouse gases). In accordance with the ATW scenario presented in Reference 3, sufficient ATWs to transmute the actinides from the existing LWRs are included in the present study. However, whereas the referenced deployment scenario includes ATWs to burn only the actinides that will be generated through 2015, this number was increased to eliminate all actinides produced by CLWRs in the present, longer-term scenario.

In a recent study at Los Alamos, a moderate-growth scenario for nuclear energy for the world (13 regions including the U.S.) was a result of an economic, energy, and environment (E^3) analysis that coupled global population estimates, productivity, and per capita energy demand to world supplies.[5] This modeling, coupled to the rest of the world, projects a growth of U.S. nuclear power to about 200 GWe in the year 2100 (this projection depends on complex interactions between many variables, including technology innovations, efficiency improvements, and carbon-taxing schemes). To supply 200 GWe by 2100, a deployment scenario was implemented that included new LWRs plus ATWs to burn the actinides produced by these LWRs. A TULWR that is fueled with 25% UO_2 and 75% ThO_2 (volume percent) would produce about 160 kg of higher actinides (mostly Pu) per year per GWe, compared to about 310 for a CLWR – this production depends strongly on reactor design (moderator to fuel ratio, refueling schedule, fuel burn-up, etc.), and may be optimised in future studies.

To transmute this LWR-produced feed, an ATW will fission about 1 100 kg of actinides per year per GWe, thus the support ratio is 7 GWe TULWRs per 1.05 GWe ATW. Therefore, in 2100, 174 GWe from TULWRs and 26 GWe from ATWs provide the 200 GWe demand. Both the retirement and deployment scenarios are shown graphically in, which includes CLWRs that retire between now and 2045, ATWs that are deployed beginning in 2020 to transmute actinides and fission products from the CLWR used fuel, TULWRs that are deployed to satisfy a growing demand for electricity, and additional ATWs that are deployed to close the back-end of the ThUOX fuel cycle.

Figure 1. **U.S. nuclear power scenario through the year 2100**



In Figure 1 the upper solid line represents a “goal” scenario for nuclear energy. With no new reactors between today and 2020 (no “evolutionary” LWR deployment), and with planned ATW and limited TULWR deployment beginning in 2020 (dashed and light solid lines), the attainable total nuclear power is computed and is shown as the solid black line. ATWs to support the growing number of TULWRs are shown as a dashed line. The numerical model attempts to “keep up” with the growth requirement.

The resultant spent fuel production, actinide feeds for ATWs, and plutonium accumulation are determined by the growth of the four types of reactors shown on this figure.

Fuel Cycle and Inventory Analysis

The various flows of materials and places they are accumulated in today’s concept for CLWRs, as a once-through, or “open” fuel cycle, are illustrated in the diagram in Figure 2. This fuel cycle has a feed and spent fuel inventory that are tracked in our calculations. The thorium-uranium oxide fuel cycle with ATWs at the backend is more complex, as shown in Figure 3. The TULWR fuel cycle is similar to the CLWR cycle, but the TULWR/ATW combined fuel cycle adds an additional resource vector for thorium to the LWR Feed “F”, a Separations Plant “S” (and inventory), the ATW process “A”, and a second, reduced-mass waste stream. The Separations plant/inventory was not included in our systems analysis, thus the feed for ATWs was taken directly from the accumulating spent fuel from CLWRs or TULWRs.

Figure 2. **Fuel cycle vectors for once-through, uranium fueled CLWRs**

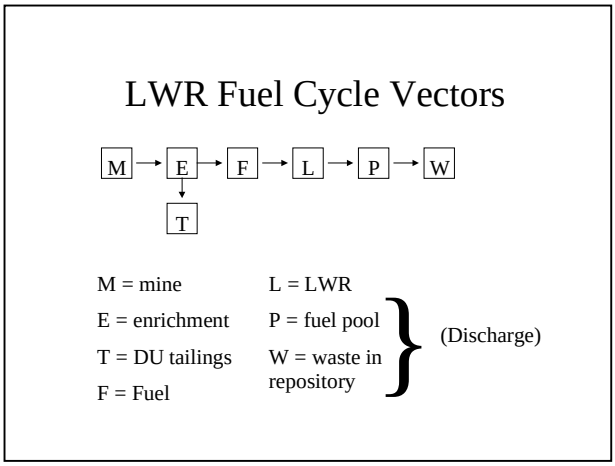
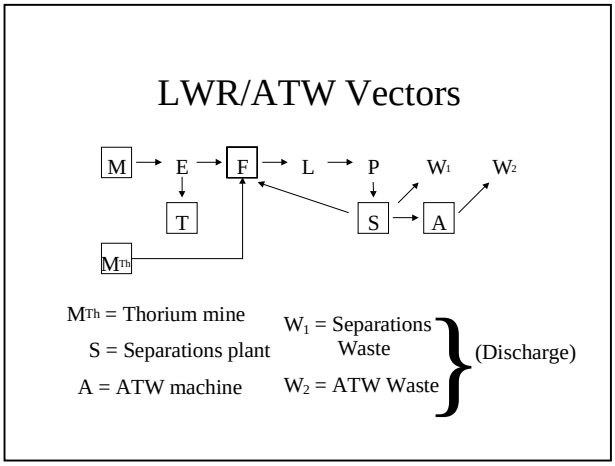


Figure 3. **Fuel cycle vectors for thorium-uranium fueled TULWRs**



A depletion analysis for a "conventional" LWR was used to provide data for calculating the fuel resource requirements, spent fuel production and accumulation, actinide transmutation, and other parameters. For CLWRs, 22 460 kg/year of 5.0%-enriched U was fed to the reactors, with annual refueling and a burnup of 50 000 MWth-days/tonne heavy metal. With this feed rate and burnup and a thermal-electric conversion of 0.325, these plants would produce a nominal 1 000 MWe (equivalent to 1 176 GWe capacity operating at a capacity factor of 0.85). To calculate natural U mining and milling required to provide this fuel, a concentration of ^{235}U of 0.0072 in natural uranium and 0.0030 in tails was used. One-group LWR cross sections and breeding/transmutation/decay chains were used to generate actinide fission, decay, and buildup during the fuel cycle.

A similar computation was done for the ThUOX fuel cycle in the TULWRs, with the same feed, efficiency, and burnup. To compute burnup and actinide production, the SAS2 driver from SCALE [6] was used to calculate new cross sections for a PWR fuel assembly with 25% U and 75% Th (volume percents), then to run a depletion analysis with ORIGEN-S. This analysis resulted in a production of just 160 kg of higher actinides per yr vs. 310 for the CLWR. For this study, a new core fueling scheme was not designed and analyzed for various safety parameters (void reactivity coefficients, Doppler effects, etc.); however, changes in the core design can significantly change feed and discharge compositions of the Th-U fuel cycle, which may decrease ^{233}U production, increase Pu production

(thus increasing total actinides), and decrease the number of TULWRs that an ATW may support. This calculation is very sensitive to spectrum-averaged cross sections, so that optimization of the reactor design, enrichment, loading, and Th:U ratio may maximize production of ^{233}U which would minimize make-up requirements for enriched uranium. Optimization could also minimize production of actinides that would require transmutation, which would allow a higher support ratio for ATWs.

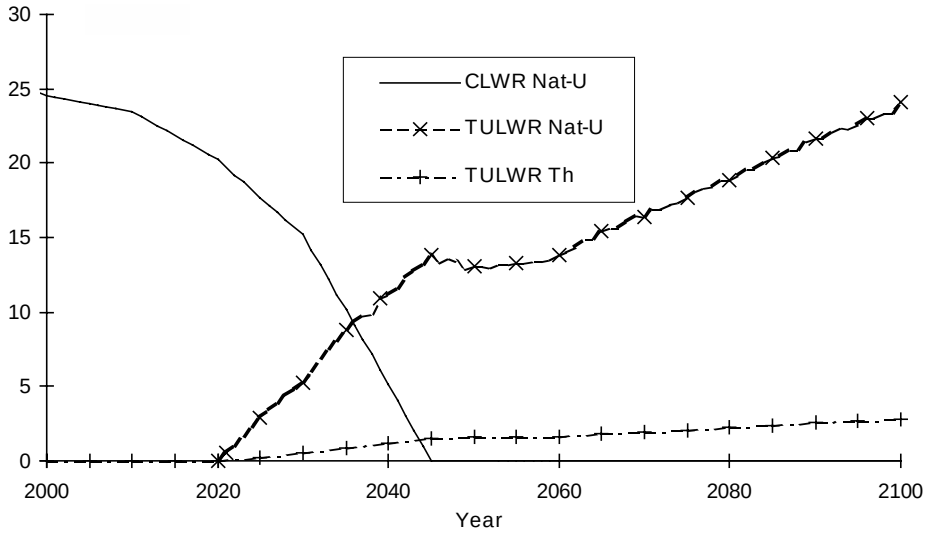
A simple spreadsheet program was used to track and accumulate the production of actinides, inventories residing in CLWR and TULWR spent fuel, and feed of actinides to ATWs (thus, loss from spent fuel pools), with values calculated by year, from 1996 to 2100. Inventories of actinides were accumulated in tables by isotope, mass was removed from the ^{241}Pu inventory as it decayed, and that mass was added to the ^{241}Am inventory. Higher actinides from CLWRs (conventional LWR) were accumulated in a single spent fuel inventory, and TULWR (LWR with ThUOX) actinides were accumulated in a separate spent fuel inventory. Actinide feed for ATWs was subtracted from LWR spent fuel inventories at a rate of 1 100 kg/yr per GWe ATW (3 000 MWth, 35% net efficiency after recirculation of power for the plant). In addition, the feed of isotopes for the ATWs was proportional to the isotopic concentrations in the CLWR and TULWR spent fuel inventories (e.g., if the TULWR pool contained 10% ^{240}Pu and 90% other actinides, then the feed for the TULWR-supporting ATW was 10% ^{240}Pu).

Finally, as a basis for calculations through the year 2100, the accumulated plutonium and higher actinide waste at the end of 1995 (the beginning of the present analysis) was based on 31 952 metric tonnes of spent fuel [7] from a computed 1422 Gwe-years of production. This would include approximately 410 metric tonnes of higher actinides from 50 GW-days/tn burnup, or more than one percent of the accumulated spent nuclear fuel from U.S. reactor operations.[8]

Resource Requirements

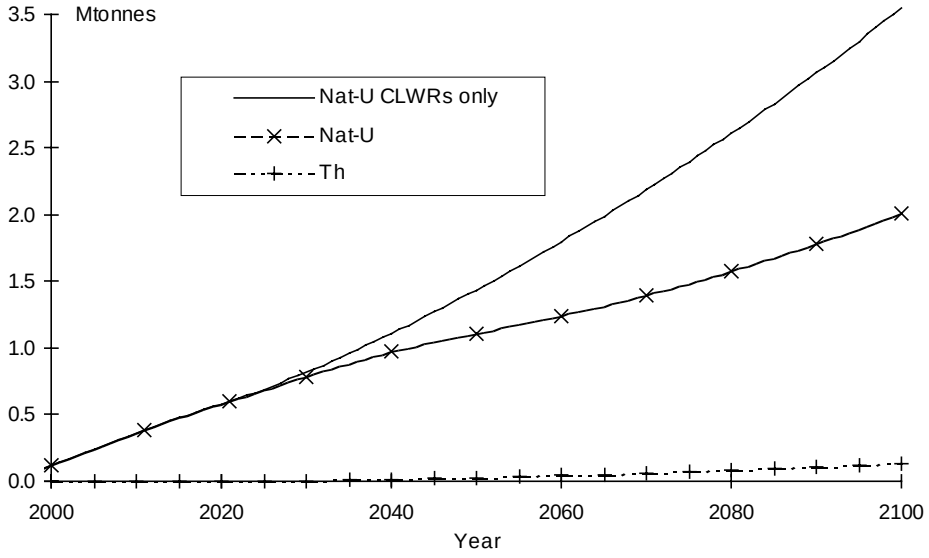
The calculations of actinide production, transmutation, and accumulation showed that the ThUOX TULWR/ATW fuel cycle can greatly change a future nuclear energy scenario, in terms of resource depletion and production and accumulation of both waste and proliferation materials.

Figure 4. **Annual resource requirements for CLWRs and TULWRs**



In Figure 4, annual resource requirements are shown for natural uranium for CLWRs and natural uranium and thorium for TULWRs that are deployed in accordance with the nuclear power scenario. The apparent fluctuation in the natural uranium requirement for TULWRs is caused by high-enrichment requirements for the first fueling of each reactor (our method assumed that no ^{233}U or Pu would be available from other sources). Each new reactor requires a much larger natural U resource its first year. In the following years, recycled ^{233}U provides some of the needed fissile composition for criticality, so less enrichment is required. This requirement for natural U could be greatly reduced with a reactor design that breeds more ^{233}U versus ^{239}Pu.

Figure 5. **Cumulative natural resource requirements for CLWRs and TULWRs**



Note: This figure includes the cumulative requirement if all nuclear energy to 2100 is supplied by once-through CLWRs only (upper line)

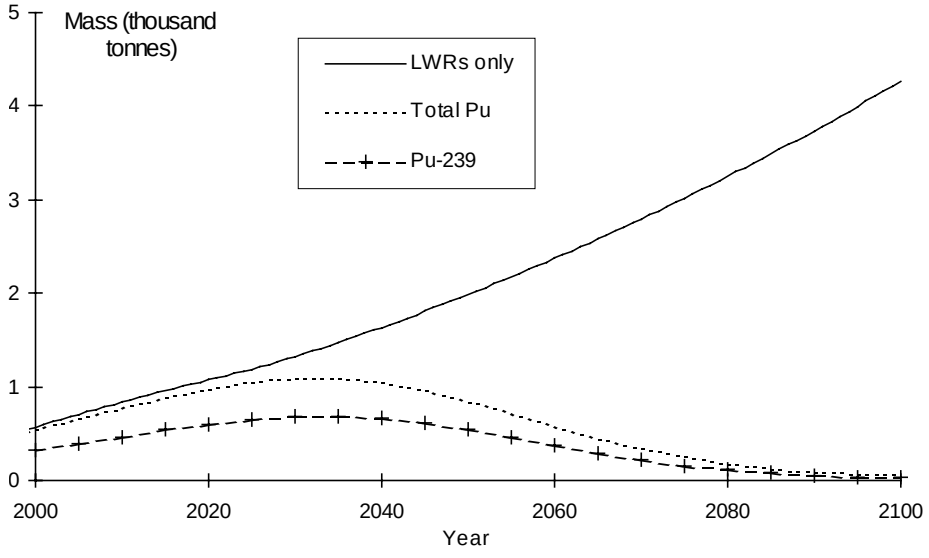
Figure 5 is a graph of the cumulative resource requirements for both natural uranium and thorium. For comparison, the figure includes cumulative uranium requirements for a scenario with only CLWRs providing growing nuclear electric production through the year 2100. Thus, if we attempt to follow this nuclear growth scenario with CLWRs alone, with no ATWs or TULWRs supplying electricity, the cumulative (U.S.) requirement for once-through LWRs for this moderate-growth scenario would almost equal the world's known (economically recoverable) uranium reserve of 4 million tonnes.[9] If this same growth scenario is applied to the global growth scenario reaching 1 000 GWe in 2100, the requirement would be 17.5 million tonnes of uranium, which is more than four times the known reserves (but about equal to the conventional resources). In contrast, the requirement with the TULWR/ATW deployment scenario is just 1.8 million tonnes of U and 0.2 million tonnes of Th (thorium is widely known to be about three times as abundant in the earth as uranium); both are well below current, known, economical world reserves.[10] A similar global deployment scenario of TULWRs and ATWs would require 9 million tonnes of U and 0.7 million tonnes of Th, which are also less than known resources. Thus, the TULWR/ATW concept, with ThUOX-fueled reactors and actinide burning, would prevent future excessive depletion of world resources. However, this analysis also demonstrates that substantial additional savings could be realized with a deployment of a reactor with higher conversion of thorium to ^{233}U.

Proliferation Attributes of TULWR/ATW Pu

Plutonium will accumulate in cooling/storage pools in spent fuel from CLWRs (currently about 30 tonnes) and from TULWRs. This plutonium is of concern not only because of the quantity accumulating worldwide, but also because of the forms of that Pu (in-core, in storage, separated, in MOX fuel, etc.), and because of the "quality" of the Pu, that is, how useful it might be for constructing nuclear explosives. Because of these concerns, the ATW program at LANL has been designed to burn U.S.-generated Pu, plus other actinides, which will both destroy the Pu and produce electricity (one current ATW concept includes thermal to electric conversion of 40%, or about 35% net for an ATW plant). Each ATW will fission about 1 100 kg of actinides per GWe produced. However, because of the accumulated inventory and ongoing production of Pu (250 kg/year/GWe), more than 900 tons of Pu will be in storage in the U.S. before deployment of the first ATW in 2020. To compute the removal of actinides from spent fuel, each ATW removed about 1000 kg/yr of actinides from the CLWR or TULWR spent fuel inventory. In Figure 6 the cumulative inventories of Pu in both TULWR and CLWR used-fuel reserves are compared. Again, if the power growth scenario represented in Figure 1 is supplied by CLWRs alone, without actinide burning or recycle, Pu will accumulate approximately like the upper curve of Figure 6, with a total inventory exceeding 3 500 tonnes by 2100. Although not of the same quality as weapons-grade Pu, and initially protected by the "reactor standard," large quantities of Pu have been identified as a significant proliferation issue.[11]

In contrast, with the TULWR/ATW deployment scenario, the Pu in inventories of spent fuel is equal to the product of the annual production (per GWe) times 1 to 5 years (the cooling time before reprocessing) times the total power of all TULWRs. All other spent fuel will be removed for processing for transmutation. The total inventory (see the dashed curve of Figure 6) would be about 67 tonnes in 2100, which is less than 2 percent of the Pu that would accumulate from the once-through fuel cycle. In addition, most of this Pu would be controlled/contained in highly radioactive fuel (less than 5-year decay time) or in processing for feed to ATWs, in contrast to current practices and plans, where spent fuel has been decaying for more than 20 years in cooling ponds, and in the future may be placed in interim storage facilities or in final, underground repositories.

Figure 6. **Pu in spent-fuel inventories resulting from an TULWR/ATW deployment scenario with 200 GWe in the year 2100**



Note: The upper solid line represents Pu inventory that would accumulate if once-through UO_2-fueled LWRs provide all the nuclear electricity.

In addition to this greatly reduced quantity of Pu in out-of-core inventories, the Pu produced in ThUOX fuel has a higher source of spontaneous neutron emission and thermal energy. Both of these "qualities" affect the attractiveness or usefulness of Pu for nuclear weapons; therefore Pu created during the scenario examined in the present study will be compared with the attributes of three different grades of Pu: Weapons-grade Pu (W-G Pu), Reactor-grade Pu (R-G Pu), and Mixed-Oxide-grade Pu (MOX-G Pu). These attributes include critical mass, thermal energy, and spontaneous neutron emissions; all of which might affect design, construction, or probability of achieving a given yield. Although criticality and yields of devices that could be created from the Pu from CLWRs and TULWRs were not examined in this study, the tables were used to examine trends and estimate differences between Weapons-grade Pu (W-G Pu), Reactor-grade Pu (R-G Pu), and the Pu that results from the ThUOX TULWR/ATW closed fuel cycle.

The thermal energy production and spontaneous neutron production or source (SNS) from the different isotopes of Pu, in per kg units, as well as the minimum mass of a given isotope of Pu that would be required to create a bare critical assembly; the Bare Critical Mass (BCM), are listed in Table 1 for each of the isotopes of Pu. While the heat produced by a quantity of Pu is influenced strongly by the 238, 240, and 242 isotopes of Pu (all α emitters), the quantity of Pu required to form a critical assembly is increased only by the 240 and 242 isotopes. Thus, the heat produced by a BCM of Pu will depend on concentrations of the ^{238} Pu, ^{240} Pu and ^{242}Pu. Each of these parameters (heating, SNS, and BCM) will be examined briefly, then products or ratios will be presented and the results interpreted as they apply to usefulness for proliferation. Isotopic compositions of Pu in the final spent-fuel pool from TULWRs is given in Table 2, along with compositions of various grades of Pu. These vales were used to compute parameters that are used to compare proliferation attributes of the different grades of plutonium, which will be presented in discussions in the following pages.

Table 1. **Spontaneous neutron emission rate (SNS), heating (computed from data from Reference 12), and critical masses (computed via SCALE, Reference 13) of bare (non-reflected) spheres (BCM) of isotopes of plutonium**

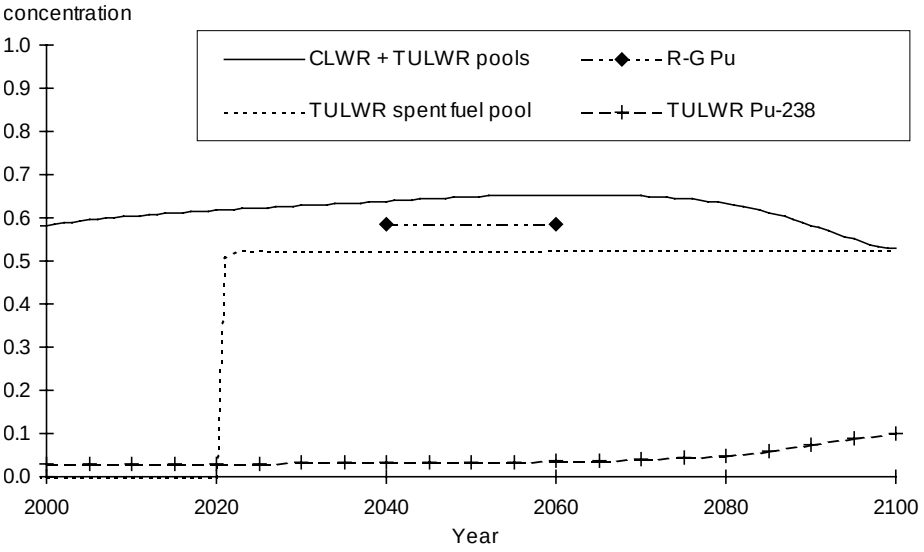
Isotope	SNS (n/s/kg)	Heat (W/kg)	BCM (kg)
Pu-238	2.60E+06	5.6E+02	10
Pu-239	2.2E+01	1.9E+00	10
Pu-240	9.1E+05	6.9E+00	36
Pu-241	4.8E+01	4.2E+00	13
Pu-242	1.7E+06	1.1E-01	92

Table 2. **Isotopic compositions (weight percentages) of five grades of plutonium: Weapons grade (W-G), Reactor grade (R-G), Mixed-Oxide Grade (MOX-G), Radkowsky Thorium seed spent Fuel (RTF), and the final TULWR plutonium in the spent fuel pool**

Isotope	W-G [14]	R-G (spent fuel) [15]	MOX-G (feed) [16]	RTF seed (spent fuel) [2]	TULWR (spent fuel)
Pu-238	0.00012	0.024	0.019	0.065	0.089
Pu-239	0.938	0.584	0.404	0.465	0.499
Pu-240	0.058	0.240	0.321	0.225	0.181
Pu-241	0.0035	0.112	0.178	0.155	0.098
Pu-242	0.00022	0.039	0.078	0.090	0.086

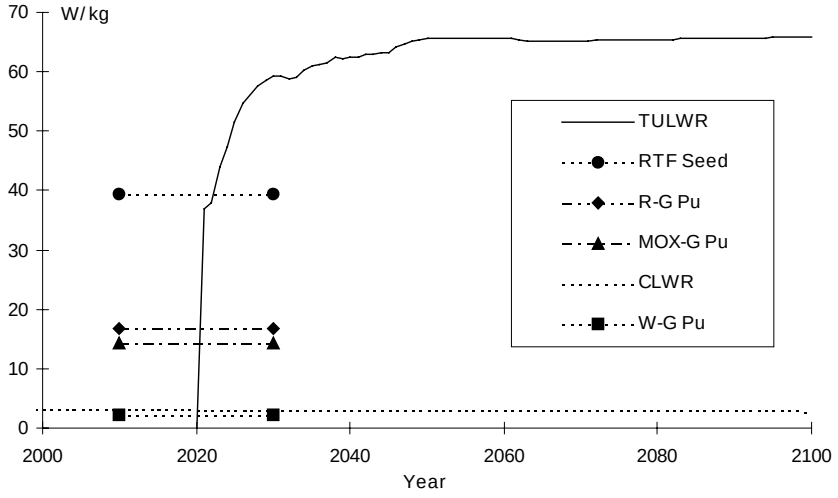
Figure 7 illustrates the fractional compositions of ^{239}Pu in the spent fuel from CLWRs and TULWRs in the scenario studied herein, as well as the increase in ^{238}Pu concentration in the combined pools. The heat that a given grade of Pu produces is simply the linear combination of isotopic heating values (W/kg, from Table 1) with the isotopic concentrations in that grade of Pu (from Table 2). Thus, a 50/50 combination of ^{238}Pu and ^{239}Pu would produce 280.95 W/kg (560/2 + 1.9/2 = 561.9/2 = 280.95), and a BCM would produce 10*280.95 or 2 810 Watts for many years (the half life of ^{238}Pu is 87.7 years). The heating per unit mass in CLWR Pu and TULWR Pu are shown in Figure 8 along with heating rates for W-G, R-G, and MOX-G Pu. For comparison with a proposed near-term ThUOX fuel cycle, the computed value for the Pu from seed-fuel elements of the Radkowsky Thorium Fuel (RTF) is also included. Again, the isotopic ratios in fed U and discharged Pu for TULWRs change with time, so the upper line fluctuates during rapid new-reactor deployment.

Figure 7. **Concentrations of ^{239}Pu and ^{238}Pu in various grades of plutonium**



Note: The upper line includes the combined Pu from both CLWRs and TULWRs. From 2000-2020 it is entirely CLWR Pu, and from 2080 to 2100 it is almost entirely TULWR Pu

Figure 8. **Heat generation in plutonium from CLWR and TULWR spent fuel**



Note: For reference, the computed value for the Pu from spent seed-fuel elements of the Radkowsky Thorium Fuel (RTF) is included.

If the bare critical masses and the isotopic concentrations in the various grades of Pu are combined linearly, as before, an index for critical mass is obtained; results are displayed in Figure 9. The Pu from the ThUOX-fueled TULWR/ATW scenario has approximately the same BCM index as the nominal R-G Pu because it has about the same concentration of ^{240}Pu and ^{242}Pu, and this BCM is much larger than the value for W-G Pu. An alternative interpretation of this figure is that the inverse of the BCM provides an indication of a relative number of weapons that could be constructed from inventories of different grades of Pu – a larger value of BCM for TULWR Pu means that fewer weapons could be created from a given mass of diverted spent fuel.

Another index of "usefulness" is produced by combining the heating value with the critical mass index, to obtain an index of alpha-decay heating per critical mass, as in Figure 10. The value shown in this figure indicates a comparison of how much heat would be generated inside comparable weapons. The value for the TULWR Pu, 1 200 Watts per Bare Critical Mass, is about 50 times the heat in a critical mass of W-G Pu, and also exceeds the heating in Pu from LWRs, that in MOX, and that in RTF spent-seed fuel. Thus any attempt to build a weapon from this material would require complex heat removal engineering solutions to prevent substantial degradation of weapon performance.

Figure 9. **Bare Critical Mass (BCM) index: the mass of a non-reflected critical assembly of a grade of plutonium**

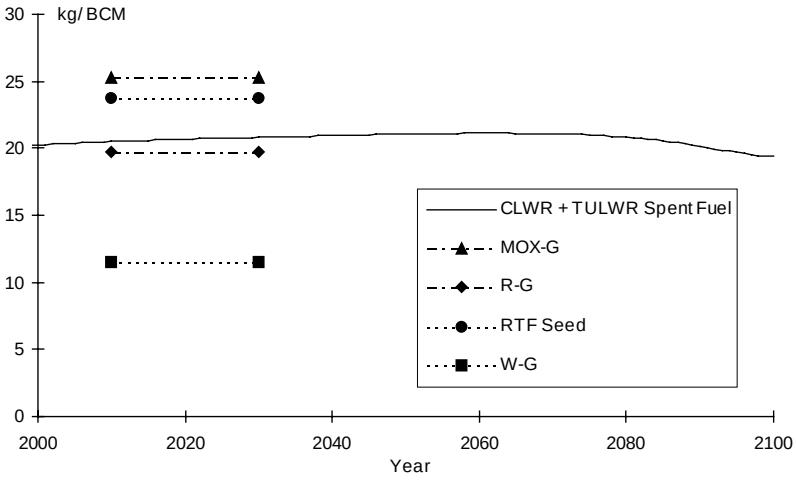
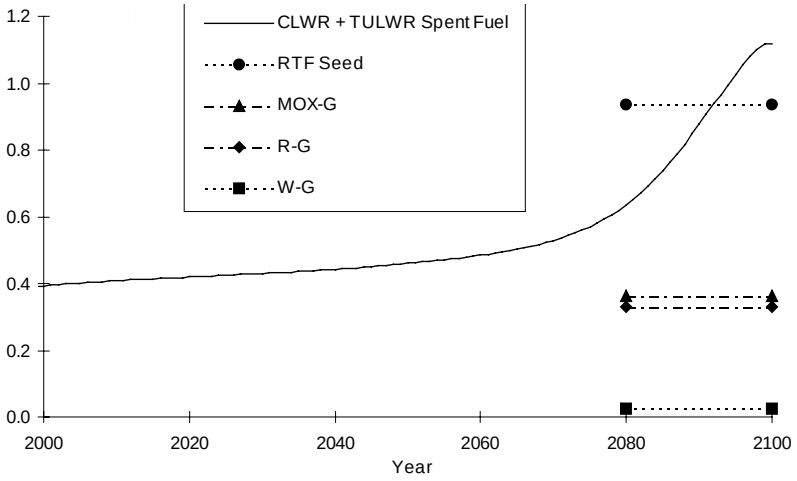
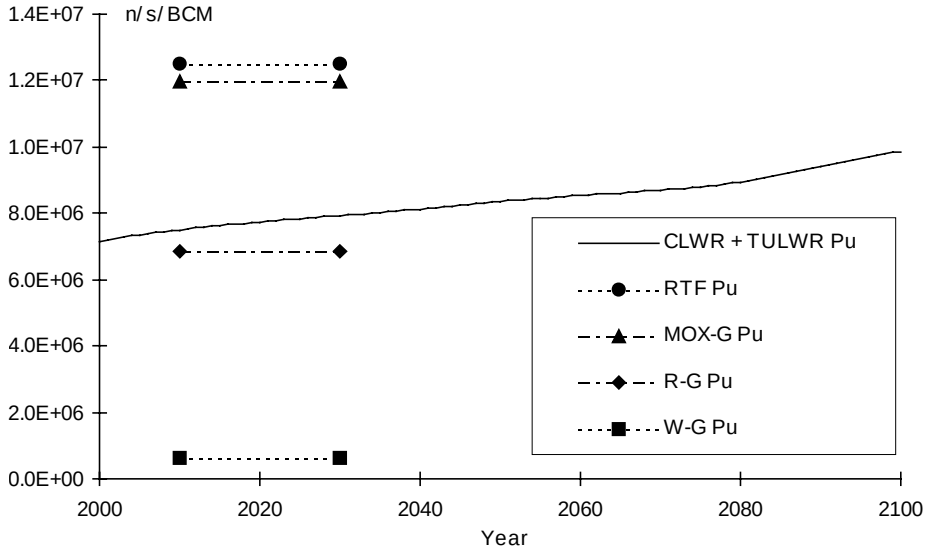


Figure 10. **Heat generation per BCM**



Note: The heat that would be generated in a bare critical sphere of Pu.

Figure 11. **Spontaneous Neutron Source (SNS) in a critical mass of various grades of Pu (n/s per BCM)**



Similar to the computation of heat/mass or heat/BCM, the SNS values from Table 1 were linearly combined with the fractional composition of Pu in each process stream to give the spontaneous neutron sources that are shown in Figure 11. The values for CLWR (left portion of the curve) and TULWR (right portion) spent-fuel Pu are both high compared with W-G Pu, and the TULWR Pu emits more neutrons per second than that from CLWRs. Note that the CLWR SNS rate increases with time because ^{241}Pu, which contributes almost nothing to the SNS, decays (to ^{241}Am) with a 14.4 year half-life, which effectively increases the concentrations of other neutron-emitting isotopes.

Fuel Cycle Cost Savings

Although a detailed analysis of fuel cycle costs was not performed for this study, other recent comparisons of once-through thorium-uranium fuel cycles with current LWR fuel cycles have predicted savings of 20 to 30 percent.[17,2] This savings excludes any additional savings that could be realized by reductions in the mass of spent fuel that requires permanent storage.

Further Studies

The results of systems analyses that were completed for this study have revealed opportunities for optimization and directions for further studies. These might include:

- Optimization of the Th-U fuel cycle to maximize conversion of thorium to fissile ^{233}U and to minimize the production of actinides and the cost imposed by the fuel-cycle-closing ATWs.
- Longer burnup fuel cycles (RTF ThUOX blanket fuel will have a burnup of 100 000 MWd/tonne heavy metal), which will also produce less actinides per unit energy, and a different mix of actinides that would require transmutation.
- Implementation of other Th-U fuel cycles, e.g., high-temperature gas reactors or fast reactors cooled with liquid lead-bismuth (the same coolant as the ATW), which would also produce a different mix of actinides and a different quality of Pu.

- Improved computational methods that include the actual histories of U.S. power plants and options for varying burnup, electrical generation efficiencies, and capacity factors in the future. This might include adding current plans for use of MOX fuel in existing LWRs or in advanced LWRs.
- Coupling of results of these studies to analyses of cost of electricity and to some form of index or prediction of proliferation risk.

Summary

A future nuclear energy scenario that includes new LWRs with thorium-oxide fuel, with follow-on processing and burning of actinides in ATWs, results in a closed fuel cycle that produces smaller quantities of Pu and other actinides than the "standard" once-through LWR fuel cycle. This TULWR/ATW offers other improvements that include reduced depletion of natural resources (50% less natural uranium), smaller volumes and less hazardous waste for disposal (98% less higher actinides), and a lesser "quality" of Pu for proliferation of nuclear weapons (50 times the heat and 15 times the rate of spontaneous neutron emission per critical mass as weapons-grade Pu). Thus, the TULWR/ATW ThUOX fuel cycle should be considered a "less-proliferative" fuel cycle. This scenario should also produce a reduction in costs associated with the fuel cycle, on the order of 20%. In addition, this scenario includes only one ATW for every seven TULWRs, and the ATW will produce electricity for sale as well as savings in terms of the forms and volume of waste for final disposal, so the ATW deployment should have a minimal impact on the cost of electricity.

Topics for further studies have been identified which may point toward opportunities of enhancing this scenario with other types of advanced reactors, fuel cycles, or processing options.

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HIGH-TEMPERATURE REACTORS:
THE DIRECT CYCLE MODULAR HELIUM REACTOR

by Michel Lecomte, Framatome

Introduction

Many projects using the HTR technology are underway in the world, notably in Japan, China, Holland and South Africa, showing that this concept is promising.

The "MHR" project described below involves an International Co-operation between General Atomics USA, MINATOM Russian Federation, FUJI Electric Japan and Framatome France.

The Modular Helium Reactor (MHR) is the result of the direct coupling a small reactor with a helium-driven gas turbine.

This package has been made possible by taking advantage of the last developments in two domains: High-Temperature Reactors; and large industrial gas turbines, magnetic bearings, and high capacity heat exchangers.

Electricity is produced with the high-temperature helium primary coolant from the reactor directly driving the gas turbine electrical generator. It encompasses high levels of passive safety while showing a promising reduction in power generating costs by increasing plant net efficiency to be a remarkable 47 per cent.

Many fuels can be used and thoroughly burnt, including plutonium from various sources.

Principles

With respect to conventional reactors, the MHR presents many interesting characteristics that can be summarised as follows:

- Direct cycle

The helium cooling the reactor is directly driving the gas turbine.

- Coated fuel particles

The fuel consists of microspheres of uranium oxycarbide cladded with layers of carbon and silicon carbide. This ceramic coating is stable and prevents significant release of radionuclides for several hundred hours even at temperatures reached in severe accidents. Furthermore, the fuel can withstand very high burnups (almost 10 times more than current reactors).

- Graphite Core

The fuel particles are contained in hexagonal graphite fuel elements.

The low power density of the core and the massive graphite core structure involve a large temperature inertia and ensure that changes in the overall core temperature take place very slowly.

- Helium coolant

The inert and ever gaseous helium coolant has several advantages. There are no neutronic reactions with the helium nor chemical or energetic reaction between coolant and fuel. The chemically inter nature of the helium also minimises interactions with other materials in the primary circuit, thereby eliminating corrosion products, thus reducing potential worker doses.

- Compactness

The MHR components are contained within two steel pressure vessels – a reactor vessel, a power conversion vessel – and a connecting crossvessel. All the plant is installed underground in a concrete silo, which serves as an independent vented low pressure containment structure. The reactor vessel is surrounded by a reactor cavity cooling system, which provides totally passive decay heat removal. The separate cooling system provides backup decay hear removal for refuelling and maintenance activities.

The power conversion vessel contains the turbomachinery consisting of two compressor sections and a turbine generator submerged in helium, and mounted on a single shaft supported by magnetic bearings. The power conversion vessel also contains three compact heat exchangers: a high efficiency recuperator, a water cooled intercooler, and a precooler.

Interesting features

Safety Characteristics

The key to achieving large safety margins in the MHR stands in controlling system temperatures and in ensuring that the integrity of the initial barrier to radionuclide release is not endangered. Additional barriers such as the containment structure provide defense-in-depth, mitigate the consequences of any release, and serve to further increase the safety margins for the MHR.

The nuclear characteristics of the graphite and low enriched uranium materials combine to produce a negative power cœfficient, meaning that the nuclear reaction is physically stopped if the core heats beyond normal operating temperatures.

The reactor is designed in such a way that even when all power supply is lost, natural heat removal will maintain the temperature of the fuel low enough to guaranty its integrity. As a result, the MHR

can withstand failure of the primary system boundary with loss of helium coolant in combination with the loss of all forced circulation without the core reaching temperatures at which significant coated fuel particle failure would occur. Radionuclide releases are significantly below regulatory limits. Thus, the combination of inherent and passive features provides large margins of safety even in the case of severe accidents.

Competitiveness

The use of inert helium gas with graphite and the refractory coated fuel allow operation of the reactor at high temperatures and generation of electricity with a remarkable 47% thermal efficiency.

The present evaluation of a commercial MHR looks promising enough to complete with other sources of electricity in this range of power. This feature makes it able to be installed in countries which do not enjoy interconnected electricity network, or far from other electrical power sources, as shown by OECD studies. On top of that let us recall the MHR is capable to produce both high quality industrial heat and electricity.

Flexibility to use fissile materials, and easy storage

Several fissile materials may be considered such as Uranium 235, combination of ^{235}U and ^{232}Th, but also Plutonium from weapons-grade material or from reprocessed commercial reactors fuel. Thanks to its high burnup capabilities, in the case of burning weapons-grade plutonium, the material balance at the end of the fuel life shows that 90% of the ^{239}Pu is destroyed, representing the highest ration to be achieved in a commercial power reactor.

With respect to the back-end of the fuel cycle, the refractory coating of the fuel continue to show a remarkable stability and inertness, as good or even better than vitrified radioactive wastes, that allow a direct storage.

The Framatome project

The main characteristics of the project are recalled in Table 1 where the most remarkable parameters are the high efficiency (47%) and the higher than usual heat sink temperature (125°C) allowing use of free rejected heat for district heating or desalination. In these cases over 80% of the nuclear energy is used.

The structure of the coated fuel particles allows high burn ups which limit the spent heavy metal volume to be handled, Table 2 illustrates that point for a case where the GT-MHR has a standard burn up of 121 GWt/t but this figure could be increased at least two folds, if the economics warrant it, to further decrease the amount of spent heavy metal.

Finally the GT-MHR is quite flexible in fuel cycles, such as Pu-Th to minimise minor actinides or pure Pu fuel in order to help decrease the weapon grade inventory of Pu in Russia or the US. Table 3 illustrates such Pu consumption on an annual basis per module. In particular around 90% of the original ^{239}Pu is consumed.

Table 1

GT-MHR Nominal Full Power Operating Parameters

Reactor Power, MWt	600
Core Inlet/Outlet Temperature, °C	491 / 850
Core Inlet/Outlet Pressures, MPa	7.07 / 7.02
Helium Mass Flow Rates, Kg/s	320
Turbine Inlet/Outlet Temperatures, °C	848 / 511
Turbine Inlet/Outlet Pressures, MPa	7.01 / 2.64
Recuperator Hot Sice Inlet/Outlet Temps, °C	511 / 125
Recuperator Cold Sice Inlet/Outlet Temps, °C	105 / 491
Net Electrical Output, MWe	284
Net Plant Efficiency, %	47

Table 2

Reactor Resource Consumption and Environmental Impact Comparison

Reactor Rating	PWR	GT-MHR	
• Thermal Power, MWt	3 914	600	
• Power Units/Plants	1	4	
• Plant Thermal Power, MWt	3 931	2 400	
• Thermal Efficiency, %	35	48	
• Plant Electrical Power, MWe	1 385	1 145	
Thermal Discharge			
• Thermal Discharge, GWt/GWe	1.8	1.1	
Equilibrium Fuel Cycle			
• Heavy Metal Loading, MT/GWt	26.8	7.5	
• Enrichment, %	4.2	15.5	(Avg)
• SWU, 10^3 kg SWU/GWe-Yr	135	221	
• Burnup, GWt-day/MTHM	47.8	121	
• U_3O_8 Consumption, MT/GWe-Yr	181	246	
Spent Fuel			
• Discharged HM, MT/GWe-Yr	21.4	5.4	
• Enrichment, % ^{235}U/Total U	0.9	4.8	(Avg)
• Discharged Pu, kg/GWe-Yr	235	109	
• Discharged ^{239}Pu, kg/GWe-Yr	171	43	

Table 3

	Feed (Kg)	Discharge (Kg)
^{239}Pu	246.3	25.6
^{240}Pu	15.7	78.1
^{241}Pu		30.5
^{242}Pu		8.7
Other Actinides		3.8
	262.0	146.7

ROUND TABLE/PANEL

FINDINGS AND CONCLUSIONS

CHAIRMAN: C.K. PARK (REPUBLIC OF KOREA)

SUMMARY RECORD OF THE ROUND TABLE[1]

Clas-Otto Wene [Chairman of Session #1]

Session #1 was essentially an introduction to the Workshop. The two papers presented aimed at setting the stage and providing a framework for investigating fuel cycle option alternatives by 2050. The future of nuclear energy will depend upon the performance of nuclear technologies and the evolution of the decision-making environment. Coming from outside the nuclear community, there will be opportunities and threats. Within the nuclear community, responses will be found based upon the strengths of the technology and the efforts that will/should be made to recognise, alleviate and/or mitigate its weaknesses. On the weaknesses and threats side, as pointed out by the speakers, we find present lack of competitiveness, absence of demonstrated, publicly accepted solutions for all radioactive waste disposal, and real or perceived proliferation and safety risks. Opportunities include a quasi-absence of greenhouse gas emissions, a mature technological development and capabilities to enhance security and diversity of supply. Market mechanisms may be seen as opportunities and challenges. Globalisation is a threat for national nuclear power programmes aiming at energy independence but it is an opportunity for financing nuclear units. Economic deregulation (or re-regulation) opens wider markets for utilities but eliminates captive markets and, thereby, increases uncertainties on future sales.

On the opportunity side, climate change concerns might be a key driver. The US presentation included an overhead showing that the present American policy on electricity production would result in CO_2 emissions around 2.5 times higher in 2020 than at the beginning of the 90s. Clearly, such a policy is difficult to reconcile with the Kyoto commitments unless there is an expectation that buying a lot of emission permits, from Russia for example, will allow not to worry about domestic CO_2 emissions. Another possible mover, although likely not as important as the first one, is security and diversity of supply, which was quoted as significant by the two speakers. Globalisation broaden opportunities for finding capitals, making financing easier as pointed out by the first paper which also stressed the benefits of deregulation in terms of potential market size.

On the threat side, nuclear waste and proliferation risks were mentioned by both speakers although the USDOE places proliferation in the first place while the NEA/IAEA paper puts waste first. Safety and costs are among key issues also. Privatisation may reduce the number of potential electricity generators eager to build and operate nuclear units. Deregulation induces more uncertainties on accessible markets and globalisation renders obsolete energy programmes aiming at maximising the use of domestic resources.

The strengths of nuclear energy include its well established technological and safety performance, the institutional framework that supports the safe construction, operation and closure of nuclear facilities.

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1 This summary was prepared by the NEA Secretariat; it aims at highlighting key issues presented by chairpersons and main points raised by the audience; it does not include the whole statement of each chairperson nor the entire discussions.

Also, the nuclear energy chain, including the entire fuel cycle, emits practically no CO_2. One the weaknesses of nuclear energy today is its economics. Although this might change with internalisation of environmental costs, security of supply premium and so on. The lack of public acceptance is very important even if there are expert consensus and engineering answers to issues raised by the public.

The bottom line seems to be that nuclear safety, proliferation risks and radioactive waste disposal are raising public concerns irrespective of the claims and demonstrations by the nuclear community that technological answers exist. These engineering answers are not adequate to convince the public, especially at a time when demand is not a strong driving factor. A key issue, in the OECD countries at least, is the need, or rather the absence of need, for additional electricity generation capacity. Why would you want a nuclear power plant, or any other power plant for what it matters, if you have no need for its outputs? Environmental issues are high on the agenda of people, the young generation in particular, but there is no doubt that the services delivered by modern technologies are highly appreciated also. When there will be a real need for new power plants, the questions will be what are the options available, what are their costs and their benefits. Renewable energy sources, such as wind and photovoltaïc, are expensive today but expected to reach competitiveness eventually with technology progress and mass production. Fossil fuel resources are limited, although physical shortages are unlikely in the foreseeable future, they have more valuable uses such as chemistry than energy production and they environmental issues such as acid rains and global warming. While nuclear energy seems outside of the potential options in many countries today, there is a possibility for a large scale renewal, a second appearance of nuclear, if and when the CO_2 commitments agreed upon in Kyoto and beyond are taken seriously.

Trevor Cook

With the stagnation of nuclear power programmes in the United States and Western Europe, in spite of some bright spots in the Far East, it will be difficult for the industry to survive. Even taking into account the fuel cycle business, few companies are likely to stay in business by 2020 and beyond. This is a real issue that will have to be overcome if and when a revival of the nuclear option will occur.

Konstantin Foskolos

Certainly it would be fantastic if we could keep all nuclear business alive as it was ten years ago. And it could be a pity to lose all this potential, this capital we have but we should not forget that if there is a demand for something the necessary capabilities can be built very quickly and very easily. The nuclear industry established itself and built its infrastructure from scratch very rapidly and successfully when large nuclear programmes were launched in the mid-70s. If we have to re-build the infrastructure again, we will have the advantage of feedback from experience accumulated. Hopefully there will be a revival of demand and the capacity will be also there in due time.

Evelyne Bertel

I don’t think that we are going to dismantle the nuclear industry even if we would have a stagnation of new orders over the next decades because we do have to live with more than 300 reactors which are in operation. Some will still be in operation for 30 years. We do have to dismantle them eventually, to manage radioactive waste and dispose of them. I’m not saying that it’s sufficient to maintain what we would like to have 20 years from now if we want to restart but there are still things going on even if we do not have new orders.

Konstantin Foskolos [Chairman of Session #2]

After having set the boundary conditions for the environment of nuclear power generation in the next 50 years, on the basis of scenarios extensively discussed in Session 1, the presentations in Session 2 attempted to describe concrete technical developments that could meet the requirements arising from such scenarios.

Three papers were presented reflecting the positions of the main actors in the domain of the nuclear fuel cycle; namely, the utilities, the industry and research organisations. As expected, while being all positive and forward-looking, these positions referred to different time horizons and set different priorities.

Utilities, particularly in the light of current market re-regulations and associated uncertainties, tend to consider rather the short and mid-term and base their strategies on the optimisation of existing installations through performance improvements of the fuel cycle that would help to reduce generation costs or – at least – would not burden too much the overall economics.

Industry, while assuring its current position in the market and reacting to more stringent environment protection requirements, focuses on preparedness for an expected revival of nuclear technology in the mid-term that would result in an optimisation of the fuel cycle including higher volumes of reprocessed fuels and a more consequent recycling of fissile materials.

According to their prospective mission, research organisations concentrate rather on long-term issues and propose technical concepts that could provide a concrete solution to the requirements for a sustainable energy supply by that time.

More specifically, the first paper authored by EDF on "Fuel Utilisation Improvements in Current Reactors" considers such improvements as an integral part of general strategies, including back-end and waste management aspects. The main objectives put forward are extended length of the fuel cycle and increased recycling of fissile materials after reprocessing. The former has an impact on reactor physics and materials behaviour which are, however, considered to be manageable issues. Nevertheless, significantly higher burn-ups than the ones currently allowed by the safety authorities will necessitate a thorough re-consideration of critical safety-related issues, which will often necessitate expensive experimental validation. The latter implies the achievement of a balance between reprocessing and MOX fabrication capacity on the one hand and the possibility to recycle MOX in the existing park of nuclear power plants. This strategy makes sense only if material resulting from reprocessing is considered to be equivalent to UO_2 from all possible viewpoints including core management, i.e., enrichment, allowed maximum burn-up, cycle length, manœuvrability and transport/logistics.

In the following discussion it became clear that, although fabrication costs for MOX are higher than for UO_2, the two options may be equivalent if one takes the whole fuel cycle, including disposal of waste, into account. It further appeared that 60 GWd/t for average burn-up represents the current limit for the author of the paper because, among other reasons, considerable design modifications (increased boron contents, burnable poison rods, etc.) would become necessary to reach higher burn-ups. On the side of recycling of reprocessed uranium, one does not expect any problems beyond the higher necessary enrichments. In parallel, CEA is working on a further reduction of enrichment tails (perhaps from presently 0.3% to 0.25%) that would provide considerable savings in resources use.

The joint paper by BNFL and COGEMA pointed out that a truly sustainable nuclear energy would require a maximisation of resource use through thorough recycling. However, recycling strategies,

including the implementation of fast (breeding) reactors, would be driven by economics and waste conditioning and disposal policies. Technical solutions for the implementation of such strategies do exist. The deployment of evolutionary reprocessing technologies would, in addition, allow to reduce unit costs by more than a factor of 1.5 if coupled with economies of scale and construction in series. By means of appropriate technical provisions, the environmental impact, already negligible today, would not grow proportionally to the installed reprocessing capacity. From the fuel cycle industry viewpoint, proliferation does not represent a real issue today. Safeguards could, nevertheless, be further improved, if required, by adequate combinations of reactor types and fuel forms and by appropriate siting (integration) of plants. Processing is thus a valuable option allowing for an optimisation of resource use, including management of plutonium stock-piles, and for adaptation to many different waste disposal options.

The subsequent discussion showed that, from a technical viewpoint, any desired degree of separation is feasible, leaving, however, open the question of associated costs. With regard to various repository options, it was pointed out that regional or international repositories may represent an ideal solution. The opinion was also expressed that regional repositories could be the only solution legally allowable in a unified Europe. The question was also raised whether an intensive discussion about reprocessing is beneficial to the nuclear cause or whether one should focus for a while on a simple nuclear technology relying on the once through cycle. However, a general consensus view was developed regarding the need to keep the recycling option alive as a major feature of a sustainable nuclear energy.

Given the strong commitment of the Japanese nuclear community to the fast reactor technology and its associated fuel cycles (more than 1 billion US$ spent per annum to this date), it was quite logical that the paper by JNC focused on potential solutions offered by the fast reactor technology to address the challenges outlined in the previous presentations. Based on a series of extensive concept studies, it was shown that transmutation of minor actinides in fast reactors is feasible without excessively affecting core performance when a 5% concentration is observed as upper limit. Multiple recycling of minor actinides (up to eight times) would help to reduce their inventory by 80%. Sophisticated fuel pin designs were proposed for fission product transmutation (duplex pellets). Attention was also paid to increased MOX enrichment in plutonium and uranium-free fuels for increased plutonium incineration. Plutonium consumption rates close to the theoretical maximums could be achieved, while introduction of UO_2 in the core blanket would help to alleviate the expected slight deterioration of reactivity coefficients.

The discussion following this presentation indicated that although a 80% reduction of the minor actinides inventory is already impressive and has no short-term drawbacks on the overall fuel cycle strategies, it would not be sufficient to gain public acceptance; in France a reduction factor between 10 and 100 is considered to be necessary for this purpose. It was also pointed out that Pu-Th fuels could have the same beneficial effects for plutonium burning as uranium-free cores or introduction of UO_2 in the blanket. It was also underlined that a further development of fast reactor would need a new, easily traceable and understandable mandate, as being 50% more expensive than the development of thermal reactors.

From that point on, the discussion became more general, addressing strategic issues. It was said that all proposals made in the papers of this session would probably lead to increased generation costs. Therefore, serious consideration should be given to cost/benefit analysis, i.e., assessing the levels of additional costs acceptable in order to obtain given achievements. For the time being not all trade-offs are easily quantifiable. To this purpose, one would need a commonly accepted basis for comparative assessment of different options. The current debate about sustainability and corresponding criteria and indicators could provide an adequate basis for such assessments. However, the completion of such an

approach is still far away. Furthermore, some aspects will remain non-quantifiable, such as the value of public acceptance, which might increase through elimination of minor actinides for example. It seems that a sound comparison of the cost and benefits of actinide transmutation versus CO_2 abatement will remain difficult for a long time. For the time being, there is no complete life cycle analysis of a scenario based on actinide transmutation, therefore, quantifying its benefits or drawbacks is not possible. However, it is an attractive option that might prove to be better than a once-through cycle. In this context, it was stressed that pursuing R&D programmes on innovative technologies is essential since present economic considerations alone are not a sufficient criteria for long-term policies.

J. N. van Geel [Chairman of Session #3]

Beyond the evolutionary concepts, the Session gave us some insights in very innovative long-term options. The evolutionary strategy would require little development; the existing LWRs, loaded with MOX fuel could reduce significantly plutonium stock piles. More ambitious strategies can be contemplated as pointed out in the three papers of the session. A revisited fast reactor concept with enhanced safety features that could not only generate electricity but help in burning actinides was described. Taking into account the feedback from experience, e.g., Phénix and Superphénix, innovative fast reactor concepts place emphasis on technical performance and safety. A key issue, nevertheless, for the future development of fast reactors will be to convince potential investors. The high temperature reactor (Gas Turbine Modular Helium Cooled Reactor) mentioned in the discussions, offers promising characteristics. The concept has been more or less abandoned after some disappointing experience years ago but seems to have regained momentum recently. In several countries, the nuclear industry is actively involved in resuming research and development on HTGRs. The potential use of thorium is an additional advantage in terms of broadening the nuclear fuel resource base. The accelerator driven systems which seems the most forward looking concepts have extremely good characteristics. The absence of critical assembly is a very favourable factor. The thorium cycle might play a role in the long term because thorium itself cannot form a critical assembly.

Regarding fuel developments, there are two trends of enhancements aiming at facilitating interim storage and final disposal of spent fuel (the once-through option) and improving reprocessing, recycling and actinide burning (the closed cycle option). The range of technological developments in this field is very broad and very rich. Each country, each company has its own type of fuel and all have their specific advantages. It might be necessary to make some choices eventually, although it is difficult to do so at this early stage in the process.

Maybe the best strategy is to wait until new ideas materialise. Hopefully, truly innovative concepts based on advanced technologies will enhance public acceptance and international consensus on viable nuclear energy options. At this stage, keeping a wide range of options open is very important but, at the same time, we should have in mind efficiency and focus our efforts on the most promising concepts and options.

Jacques Porta

I think that we can work for the medium, long and very long term and develop very innovative advanced concepts besides accelerator driven systems. CEA in particular is investigating strategies to cope with plutonium stock files, in the French context. There are technical solutions to reach an equilibrium (between plutonium generation and plutonium burning) by 2020-2030, assuming that one third of the French reactors can burn plutonium. The next step would be to deal with plutonium

inventories accumulated since the beginning of the French nuclear programme. Dedicated reactors, e.g., CAPRA, could be used for this purpose. Ultimately, if nuclear energy continues to be used in the very long term, to 2050 and beyond, it seems inevitable to come back to the breeder concept having in mind the limits of uranium resources.

Hubert Rouyer

I would like to offer some comments on the issues raised and the solutions at hand. The papers presented have showed that technical solutions exist to address issues such as proliferation and safeguards, plutonium and actinide accumulation. Technical solutions are not perfect and work is continuing in most field to enhance the existing approaches and/or find more efficient ones, but we do have solutions. However, this is not the main point. The key question is: are the solutions economically viable? And there is no clear cut answer since it is very difficult to predict economic conditions that will prevail by 2030-2050. It seems likely that fossil fuel resources will eventually become more scarce and, thereby, more expensive. Consequently, the competitive margin of nuclear energy is likely to increase in the long term and innovative technologies that are too expensive today might become viable in the long term. Moreover, the present R&D programmes on plutonium management, actinide separation and transmutation, enhanced reactor concepts and reprocessing processes will contribute to cost reduction as well as technological improvements. Therefore, I am confident that by 2050 we will have technical solutions economically viable. Maybe by that time we will be able to eliminate plutonium and actinides without cost penalty.

C. K. Parks [Chairman of the Workshop]

The presentations and discussions have raised a number of issues and its seems to me that it is too early to draw conclusions from the Workshop. We started yesterday by a session introducing the rationale for and issues raised by a 1 000 GWe scenario in 2050. The following sessions provided scientific and technical information on present status and future prospects regarding reactors and fuel cycle options. We have discussed extensively our preparedness for a scenario leading to a 1 000 GWe nuclear capacity by 2050. Maybe we end up with more questions than answers but I think that the exchange was useful and created a real synergy between diverse viewpoints. Too that extent, the Workshop can be considered as successful.

However, mainly owing to the lack of time, we were not able to cover the full spectrum of relevant technical activities which are currently underway or expected to be carried out in the future. Moreover, we have not covered adequately other non-technical issues that are equally important from a strategic viewpoint. It seems to me that the non-technical issues might be more decisive and be driving factors for the future of nuclear energy. I would hope, therefore, that a similar forum could be provided by the NEA to continue the dialogue on those issues and help enhancing our understanding of the opportunities and challenges of a 1 000 GWe nuclear scenario by 2050.

Before declaring the Workshop closed, I would like to thank the speakers for their excellent presentations and the audience for its active participation and valuable comments.

LIST OF PARTICIPANTS

BELGIUM

Mr. Paul Havard	ELECTRABEL
Mr. Gabriel Michaux	Ministère des affaires économiques

CZECH REPUBLIC

Dr. Miloslav Hron	Nuclear Research Institute Rez plc

FRANCE

Mr. Luc Chaudon	CEA
Mr. Denis Deroubaix	Cogema – DSI/MA
Mr. Jean-Paul Grouiller	CEA
Mr. Alain Languille	CEA
Mr. Yves Lapierre	CEA
Mr. Michel Lecomte	Framatome
Mr. Ph. Martin	CEA
Mr. Dominique Ochem	CEA
Mr. Jacques Porta	CEA
Mr. Jean-Luc Provost	EdF/Gdf
Mr. Hubert Rouyer	CEA
Mr. Bruno Sicard	CEA

GERMANY

Dr. Helmut D. Fuchs	GNS Gesellschaft für Nuklear-Service mbH

JAPAN

Dr. Toru Ogawa	JAERI
Mr. Hiroyuki Tsuchi	Tokyo Electric Power Company (TEPCO)
Dr. Toshio Wakabayashi	Power Reactor and Nuclear Fuel Development Corporation

KOREA

Dr. C.K. Park	Korea Atomic Energy Research Institute
Mr. Tae-Yoon Eom	Korea Atomic Energy Research Institute

THE NETHERLANDS

Prof. J.N. Van Geel	Institute for Transuranium Elements
Mr. G.C. Van Uitert	Ministry of Economic Affairs

RUSSIAN FEDERATION

Dr. Konstantin O. Mikitouk	Kurchatov Institute

SWEDEN

Prof. Clas-Otto Wene	Institutionen for Energiteknik

SWITZERLAND

Mr. Konstantin Foskolos	Paul Scherrer Institute

UNITED KINGDOM

Mr. Ray Dodds	British Nuclear Fuel plc.
Dr. Allan Duncan	Environment Agency
Mr. Mike Dunn	British Nuclear Fuels plc.
Mr. Robert Gunn	Department of Trade and Industry
Dr. Peter Parkes	British Nuclear Fuels plc.

UNITED STATES

Mr. Trevor Cook	US Department of Energy
Dr. Francesco Venneri	Los Alamos National Laboratory

INTERNATIONAL ORGANISATIONS

European Commission	Mr. Hans Forsström
International Atomic Energy Agency	Dr. Kosaku Fukuda
	Mr. Marc Giroux
OECD Nuclear Energy Agency	Mr. Philippe Savelli
	Mr. Claes Nordborg
	Mrs. Evelyne Bertel

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