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Actinide and Fission Product Partitioning and Transmutation

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and the
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NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the 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 28 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

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

FOREWORD

Over the past decade, interest in partitioning and transmutation (P&T) has grown in many countries around the world with the expectation that P&T could be a key technology to reduce the need for geological disposal of spent fuel and high-level waste. In order to give experts a forum to present and discuss state-of-the-art developments in the P&T field, the OECD/NEA has been holding biennial Information Exchange Meetings on Actinide and Fission Product Partitioning and Transmutation since 1990. The meetings have been held in Mito (Japan) in 1990, at ANL (USA) in 1992, in Cadarache (France) in 1994, in Mito (Japan) in 1996, in Mol (Belgium) in 1998, in Madrid (Spain) in 2000 and in Jeju (Korea) in 2002. These meetings are co-organised by the NEA Secretariat and major laboratories in member countries, and have often been co-sponsored by the European Commission (EC) and the International Atomic Energy Agency (IAEA). The 8th Information Exchange Meeting was held in Las Vegas, Nevada, USA on 9-11 November 2004 and organised in co-operation with the US Department of Energy, the EC and the IAEA.

The information exchange meetings on P&T form an integral part of NEA activities in the field of advanced nuclear fuel cycles. An overview of NEA activities on P&T and relevant publications are available at http://www.nea.fr.html/pt/welcome.html.

The scope of the information exchange meetings covers all of the major scientific topics related to P&T. The main theme of the 8th Information Exchange Meeting was the impact of P&T on radioactive waste management.

These proceedings include all of the papers presented at the 8th Information Exchange Meeting. The papers presented in the general session are provided in printed form, whereas others papers and posters can be found in the enclosed CD-ROM. The opinions expressed are those of the authors only, and do not necessarily reflect the views of the NEA, any national authority or any other international organisation. These proceedings are published on the responsibility of the Secretary-General of the OECD.

Acknowledgements

The OECD Nuclear Energy Agency (NEA) gratefully acknowledges the University of Nevada Las Vegas for hosting the 8th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation. The Agency also extends its gratitude to the US Department of Energy, the European Commission (EC) and the International Atomic Energy Agency (IAEA) for their co-operation.

TABLE OF CONTENTS

TECHNICAL SESSION SUMMARIES

The following papers are only available in electronic form on the CD-ROM accompanying this booklet:

SESSION I: P&T SYSTEMS AND WASTE MANAGEMENT

Chairs: Philippe Finck (ANL, USA) and Abraham Van Luik (DOE, USA)

SESSION II: PARTITIONING TECHNOLOGY

Chairs: Jae Hyung Yoo (KAERI, Korea) and Jan Uhlir (NRI, Czech Republic)

SESSION III : FUELS FOR TRANSMUTATION DEVICES

Chairs: Jan Wallenius (KTH, Sweden) and Jean-Paul Glatz (ITU, Germany)

POSTER SESSION I : P&T SYSTEMS, WASTE MANAGEMENT, PARTITIONING TECHNOLOGY AND FUELS

Chairs: Dominique Warin (CEA) and Joachim Knebel (FZK)

SESSION IV: TRANSMUTATION - GENERAL

Chairs: Frédérique Varaine (CEA, France) and Massimo Salvatores (CEA, France)

SESSION V: TRANSMUTATION - ADS

Chairs: Pierre D'Hondt (SCK-CEN, Belgium) and Hiroyuki Oigawa (JAERI, Japan)

POSTER SESSION II: TRANSMUTATION

Chairs: Enrique Gonzalez (CIEMAT) and Stefano Monti (ENEA)

EXECUTIVE SUMMARY

The OECD Nuclear Energy Agency (NEA) has, in response to interest from member countries, an ongoing activity on Partitioning and Transmutation (P&T) of nuclear waste. One of the key components of this international activity is a series of Information Exchange meetings, providing experts with a forum to present and discuss current developments in the field.

This Information Exchange Meeting was the eighth in the series. It was hosted by the University of Nevada, Las Vegas, USA and was organised in co-operation with the US Department of Energy, the European Commission and the International Atomic Energy Agency. Earlier meetings have been organised in Mito, Japan (1990), at ANL, USA (1992), in Cadarache, France (1994), in Mito, Japan (1996), in Mol, Belgium (1998), in Madrid, Spain (2000), and in Jeju, Korea (2002).

The general theme of the 8th Information Exchange Meeting was the impact of advanced fuel cycles to waste management strategies.

The meeting was opened with a general session, in which six national and two international P&T programmes were presented, covering current activities in Japan, the United States, France, the Republic of Korea, the Russian Federation and China, as well as the international programmes of the European Commission (EC) and of the International Atomic Energy Agency (IAEA).

Five technical sessions and two poster sessions, covering mainly scientific and technical issues in the P&T field, were organised. The contents of the different sessions are described below.

Session I: P&T Systems and Waste Management

Session I comprised six papers in total, providing information on the potential impacts of P&T on nuclear waste repositories and on the preliminarily evaluations of the P&T costs as well as the qualified waste streams generated. It was noted that one of the benefits of P&T for waste repositories was a delay or reduced need for additional repository capacity.

Session II: Partitioning Technology

Five papers provided information on recent progress in aqueous and pyrochemical processing of spent fuel, conversion of oxide into a metal or chloride for a pyrometallurgical partitioning, electrochemical separation of actinide and lanthanide in a molten fluoride media and experience of the crush-leach process for treating used TRISO-coated fuels. Recent studies of the partitioning of cesium and strontium, aiming at reducing the heat load of spent nuclear fuels in a geological repository, were also discussed.

Session III: Fuels for Transmutation Devices,

Six papers were presented, covering research on nitride fuel and pyrochemical process developments, preliminary performance analysis of the metallic fuel for ADS, characterisation of actinide alloys as fuels, modelling, fabrication, characterisation and irradiation of uranium free nitride fuel, and conversion of reprocessed plutonium and neptunium into oxides form using direct denitration. Design concepts and process analysis for fuel manufacturing was discussed in one of the presentations.

Session IV: Transmutation; General

Session IV included five papers. An overview of experiments to be performed in the PHENIX reactor in France was given. A concept of a pebble bed HTR in a once-through fuel cycle mode to reduce the radiotoxicity of minor actinides produced in LWRs was introduced, followed by an inter-comparison of two computational fluid dynamics codes. The last two papers dealt with nuclear data issues, including a complete evaluation of Pb and Bi isotopes, and a proposal for a comprehensive experimental programme covering most minor actinide nuclear data in a wide energy range.

Session V: Transmutation; ADS

Session V consisted of six papers. The session started with an overview of the European research programme for the transmutation of high-level nuclear waste in an accelerator-driven system. Next, the results of the transmutation performance assessment in the small-scale ADS was presented, followed by a paper on R&D activities for accelerator-driven transmutation system. An overview of the experimental reactor-accelerator coupling project illustrated the ADS related activities in the USA. Safety issues and safety indicators for accelerator-driven transmuters with dedicated oxide fuels were discussed. Finally, a paper on neutronic analysis studies of the spallation target window for a gas-cooled ADS concept was presented.

Poster Sessions I and II

The poster sessions included 38 papers, covering a large range of topics, such as the effect of P&T on waste management strategies, thermodynamic data of actinides, fuel behaviour, waste immobilisation and deactivation, and separation process development, transmutation of actinides, nuclear data measurements, materials for ADS systems and target development.

In the Closing Session, the session chairs presented the highlights of their respective sessions. The chairs of Session I (A. Van Luik and P. Finck) stated that there was evidence of the P&T community becoming fully integrated with its users what concerned the back end of the fuel cycle. They agreed with the suggestions that the discussions of the back-end of the fuel cycle should be included in the programme of future P&T Information Exchange Meetings. In addition, the chairs of Session I recommend that waste management organisations seriously look at the request for safety evaluations in support of the on-going NEA study on the effect of advanced fuel cycles on waste management policies.

The chairs of Session IV (M. Salvatores and F. Varaine) remarked that, at this conference, relatively few oral presentations were made in the crucial fields of materials and heavy liquid metals (HLM) technology, and it was recommended to take this point into account in future workshops.

The chairs of Session V (P. D'Hondt and H. Oigawa) pointed out that a number of technical challenges are common to different concepts, such as safety, material damage by protons and neutrons, thermal-hydraulics, corrosion, etc. These are suitable topics for international collaboration, in particular the basic ideas for safety evaluation of ADS. Experiments related to ADS, performed at universities, are also very welcome to foster the involvement of the next generation's nuclear scientists and engineers.

The scientific chair of the meeting, James Laidler, closed the meeting. The next (9^{th}) Information Exchange Meeting on P&T is scheduled to be held in Nîmes, France in autumn 2006.

WELCOME ADDRESSES

Shane Johnson
Office of Nuclear Energy, Science and Technology, USA

No paper was available at the time of publication.

Paul Ferguson
Vice President for Research and Graduate Studies
University of Nevada, Las Vegas

On behalf of the University administration, faculty, staff, and students, I would like to welcome our guests from the international transmutation research community and from the national advanced fuel cycle initiative.

We are very pleased to be the host site of this international meeting and we hope you find our facilities accommodating.

The issue of what to do with our nation’s nuclear waste has become a significant issue for the citizens of Nevada. Our Congressional representatives from the State of Nevada have their eyes and ears focused on alternate approaches to nuclear waste mitigation, now more than ever, with recent decisions regarding a potential repository at Yucca Mountain. I believe I can say without doubt that the transmutation of high-level radioactive waste is a technology that is important to our representatives and our community, because it provides a significant reduction in radiological hazard and has the potential to remove nuclear waste from Yucca Mountain. This is, of course, a key issue with our Congressional representatives who have been ardent supporters of funding the national Transmutation programme.

We believe that University of Nevada, Las Vegas is well-situated to lead the academic effort through the continued development of the research programme at UNLV and collaborations with other universities and research entities. As part of that commitment to investigating a key problem facing our community, UNLV began its research programme, the Transmutation Research Programme, to participate in the development of partitioning and transmutation technology as a part of the national effort. Now it it’s fourth year, this programme has become one of the most exciting new science and engineering research projects at UNLV.

The programme currently supports 23 faculty-supervised graduate student projects involving 36 graduate students and 25 faculty members in six academic departments across the UNLV scientific and engineering communities, with research tasks spanning the range of technology areas for transmutation, including chemical separation of uranium from spent nuclear fuel, methods of fuel fabrication, optimisation of super-conducting components for proton accelerators, and corrosion of materials exposed to lead-bismuth eutectic.

I trust you will all have a productive meeting and enjoy your visit to Las Vegas.

Thierry Dujardin
OECD/NEA Deputy Director, Science and Development

On behalf of the OECD Nuclear Energy Agency, it is a great privilege and my pleasure to welcome you all, this morning, to the 8th Information Exchange Meeting on Actinide and Fission Products Partitioning and Transmutation. At the outset of this meeting, I have three very pleasant tasks to perform.

Firstly, on my own behalf and that of everyone present, I would like to thank our hosts, Dr. Anthony Hechanova and Dr. Kathleen Lauckner from the University of Nevada Las Vegas for welcoming us in Las Vegas and for hosting this meeting in such a magnificent university campus. Outside of the USA, it is sometimes difficult to convince people that you are going to Las Vegas for work but I am convinced that this superb campus and its surroundings will be very stimulating for all our discussions. Thank you for the great job you made for the preparation of our meeting and for your hospitality. I would also like to extend my thanks to the US Department of Energy and to its Office of Nuclear Energy, Science and Technology for their contribution to the organisation of this meeting and more broadly for their continuous and strong support to the work of the OECD Nuclear Energy Agency.

Las Vegas is joining the already well-known list of cities hosting OECD/NEA Information Exchange Meetings which form part of the broader Information Exchange Programme on Actinide and Fission Product Partitioning and Transmutation. Let me remind you of the first meeting held in Mito City (Japan) in 1990, followed by biennal meetings held at Argonne National Laboratory (USA), in Cadarache (France), back to Mito City (Japan), then Mol (Belgium), Madrid (Spain) and most recently in Jeju Island (Republic of Korea) in 2002.

Secondly, I would like to thank the members of the Scientific Committee and especially its Chair, Dr. James Laidler, from the Argonne National Laboratory. Their work was essential for the fruitfulness of this meeting. Thanks to Jim's involvement in the US programme and in most of NEA relevant activities, we were in good hands.

Thirdly, the NEA is also very pleased to have co-operated with the European Commission and the International Atomic Energy Agency which are both co-sponring this meeting. I am particularly pleased to welcome Messrs. Ved Bhatnagar and Alexander Stanculescu who represent the European Commission and the IAEA, respectively.

As in previous OECD/NEA meetings, I would also like to acknowledge particularly the participation of non OECD countries. I have noted with great interest the increasing number of valuable contributions from these countries. We are pleased to welcome them here to enhance co-operation between all countries interested in partitioning and transmutation with a view to best serving the interest of all.

Most of you are already aware that the objective of the OECD/NEA Information Exchange Programme on Actinide and Fission Product Partitioning and Transmutation, established 15 years ago, is to enhance the value of basic research in this area by facilitating the exchange of information and discussions of programmes, experimental procedures and results. This Programme was established under the auspices of the NEA Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle (Nuclear Development Committee, NDC) and is now jointly co-ordinated with the NEA Nuclear Science Committee (NSC).

The meetings, part of this Information Exchange Programme, are intended to provide a biennial review of the state of the art of partitioning and transmutation. They are co-organised by the NEA Secretariat and major laboratories in member countries.

I have not enough time to enter into the details of the NEA activities in the P&T field and some of them will anyway be presented during this meeting. However let me focus on one point with a broader perspective.

The recent months may be looked at as a turning point in the consideration of nuclear energy as a viable option for future sustainable energy policy. It was very well emphasised at the last World Energy Congress in Sydney, last September, which first conclusion was that the world would need all sources of energy and that no energy source should be idolised or demonised. The main reasons for this nuclear comeback on the world energy agenda are well-known: increasing concerns about the security of energy supply and climate change. However, there is still a gap between a comeback on the agenda and an effective and efficient role of nuclear energy in the future world energy mix. As the Generation IV International Forum stated it, future nuclear energy systems will require better economics, increased safety and reliability, enhanced resistance to proliferation and improved physical protection, and a better management of waste. In many OECD countries, the later goal is the key issue for public acceptance.

With its potential of strongly reducing the amount and the radiotoxicity of waste ultimately disposed of, partitioning and transmutation and advanced fuel cycles offer opportunities which should be considered among futre options to increase the public confidence in the safe and sustainable management of high-level radioactive waste. Today, from a NEA viewpoint, this seems very well understood: other committees than the NDC or the NSC have activities linked to P&T. For instance, the Radioactive Waste Management Committee has organised a topical session on the potential impacts of a P&T programme on the safety of a waste repository. They concluded that it would be necessary to perform comprehensive systems studies without restricting the considerations to waste inventories and heat production. Social concerns and public confidence were behind these conclusions.

In this regard, the recent changes of the mandate and name of the former NEA Working Party on scientific issues in Partitioning and Transmutation (WPPT) to the new Working Party on scientific issues of the Fuel Cycle (WPFC) should not be seen as a move away from P&T, but as the recognition that P&T is now at the heart of long-term nuclear fuel cycle policies. The list of current and future studies of this Working Party demonstrates this: lead-bismuth eutectic technology, flowsheet studies, separation criteria, fuel cycle transition scenarios, etc. Some of these studies will strongly benefit of the results of the NDC Expert Group on the *Impact of Advanced Fuel Cycle Options on Waste Management Policies* to be completed early next year.

To highlight the increasing awareness of the importance that P&T may have for future long-term nuclear policy, let me quote an excerpt from a letter sent last month by the US Secretary of Energy, Spencer Abraham, to the NEA Director General on the occasion of the 50th meeting of the OECD/NEA Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle. After underlining that the Committe has undertaken forward-looking projects to establish consensus views of its member governments, the Secretary wrote:

"The impartiality and insights embodied in these views have been very useful to the United States Government in establishing policies regarding the civil use of nuclear energy. Technical areas where Nuclear Development Committee reports have been particularly useful to the United States include…

Sustained efforts to document the status and prospects of radionuclide partitioning and transmutation through specific studies and the Committee's support of a longstanding series of workshops on this subject;…"

I have no better evidence of the relevance and the usefulness of our meeting.

Before concluding, I am glad to announce that the NEA received a proposal from the CEA, the French Atomic Energy Commission, to host the 9th Information Exchange Meeting. This proposal was formally approved last month by both the NSC and NDC. The 9th IEM will be held in Nîmes, in the south of France, in 2006 (end of September/beginning of October). Technical visits to the Phenix reactor, where transmutation experiments are conducted and to Atalante where partitioning studies are carried out, will be organised. Both Phenix and Atalante are on the CEA Marcoule site near Nîmes. I personally take as a clear signal of the dynamism of the P&T activities worldwide that the follow-up to this Las Vegas meeting is already decided at the outset of the meeting itself.

Ladies and Gentlemen, I am looking forward to the presentations and discussions during the next three days. I wish you a fruitful and rewarding meeting and an enjoyable stay in Las Vegas. Thank you for your kind attention.

GENERAL SESSION

National and International Programmes on P&T

Chairs: James Laidler (ANL, USA) and Thierry Dujardin (OECD/NEA)

RECENT RESEARCH AND DEVELOPMENT ACTIVITIES ON PARTITIONING AND TRANSMUTATION OF RADIOACTIVE NUCLIDES IN JAPAN

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Abstract

In Japan, R&D activities for partitioning and transmutation (P&T) have been promoted under the OMEGA programme for more than 15 years. These activities were reviewed by the Atomic Energy Commission in Japan in 2000. In accordance with the results of the review, three institutes, the Japan Atomic Energy Research Institute (JAERI), the Japan Nuclear Cycle Development Institute (JNC) and the Central Research Institute of Electric Power Industry (CRIEPI), are continuing the R&D on the P&T technology. This report summarises the recent activities in Japan by these institutes. JAERI is engaging in the R&D on the Double-strata Fuel Cycle concept consisting of the partitioning process of the high-level waste and the dedicated transmutation cycle using the accelerator driven system (ADS) fuelled with the minor actinide (MA) nitride fuel. JNC and CRIEPI are engaging in the R&D on the P&T technology using commercialised fast reactors (FR), where JNC is mainly in charge of the MOX fuel and the aqueous reprocessing, while CRIEPI is mainly in charge of the metallic fuel and the dry reprocessing. The R&D activities on FR are organised under the Feasibility Study on Commercialised Fast Reactor Cycle Systems.

Introduction

In Japan, the spent fuel discharged from nuclear reactors is to be reprocessed, taking account of the low self-sufficient rate of the energy resources. To fulfil the nuclear fuel cycle, the disposal of the high-level radioactive wastes (HLW) discharged from the reprocessing plant should be steadily implemented. In Japan, the legal framework and the implementation entities were established around the year of 2000.

In parallel to such a movement, the Japanese Government has promoted the research and development (R&D) activities for partitioning and transmutation (P&T) of radioactive nuclides under the long-term R&D programme called OMEGA for more than 15 years, aiming at the reduction of the burden of the backend of the nuclear fuel cycle. These activities were reviewed by the Atomic Energy Commission in Japan in 2000. In accordance with the results of this review, three main institutes in Japan are continuing the R&D on the P&T technology. The Japan Atomic Energy Research Institute (JAERI) is engaging in the R&D on the Double-strata Fuel Cycle concept consisting of the partitioning process of the high-level waste discharged from reprocessing plant and the dedicated transmutation process using the accelerator driven system (ADS) fuelled with the minor actinide (MA) nitride fuel. The Japan Nuclear Cycle Development Institute (JNC) and the Central Research Institute of Electric Power Industry (CRIEPI) are engaging in the R&D on the P&T technology using commercialised fast reactors (FR), where JNC is mainly in charge of the MOX fuel and the aqueous reprocessing, and CRIEPI is mainly in charge of the metallic fuel and the dry reprocessing. This report overviews the recent R&D activities by these institutes.

Activities in JAERI

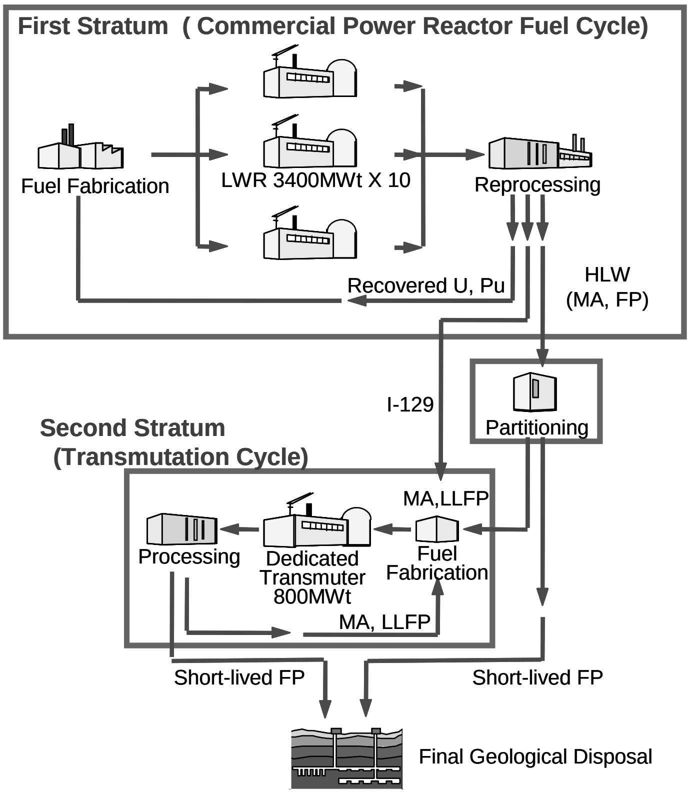
JAERI has proposed the P&T technology based on “Double-strata Fuel Cycle Concept” [1]. Figure 1 shows a schematic outline of this concept. A dedicated transmutation fuel cycle, i.e. the “second stratum”, is established separately from the commercial power generation fuel cycle, i.e. the “first stratum”, where the plutonium is recycled. The HLW exhausted from the reprocessing plant in the first stratum is treated at the partitioning plant and separated into four groups: transuranic elements (TRU), Tc and noble metal, Sr-Cs, and the other elements. The TRU mainly consisting of minor actinides (MA), namely Np, Am and Cm, is the principal object of the transmutation because these elements dominate the potential radio-toxicity in the HLW for a long term later than hundreds years after the reprocessing.

In the transmutation fuel cycle, nuclear fuel mainly consisting of MA is used to enhance the transmutation efficiency. Using such MA fuel, the critical reactor would encounter some difficulties in its safety and controllability aspects. The accelerator-driven subcritical system (ADS) is, therefore, selected as the first candidate of the dedicated transmutation system in the Double-strata Fuel Cycle Concept.

There are, however, several technical issues to be solved for the realisation of a large-scale ADS. In JAERI, therefore, various research and development (R&D) activities on the ADS are being conducted in the field of the proton accelerator, lead-bismuth eutectic (LBE or Pb-Bi) and subcritical reactor physics as well as the conceptual design study of a large-scale ADS.

It is also important to evaluate the impact of the P&T technology on the waste disposal concept. The preliminary results of the evaluation are also briefly mentioned afterwards.

Figure 1. **Double-strata fuel cycle concept in JAERI**

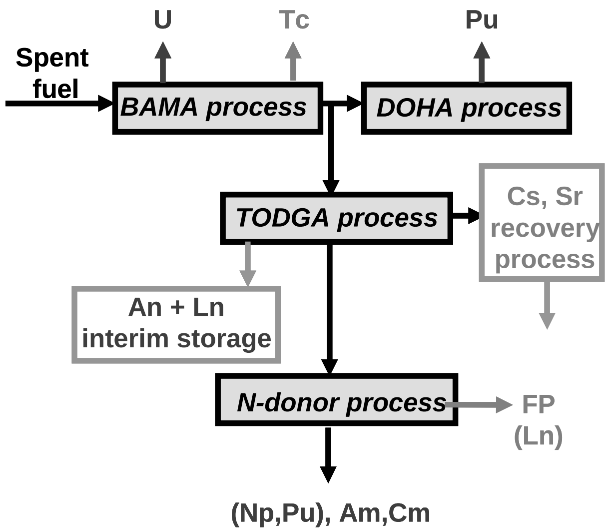


R&D on partitioning process in JAERI

After the establishment of the "4-group Partitioning Process Concept", the basic experimental confirmation was implemented by using condensed high-level radioactive liquid waste [2]. The size of the experiment was about 1/1000 of the expected commercial plant. It was confirmed that Am, Cm, the platinum group, Sr and Cs can be separated as expected.

Although the 4-group Partitioning Process is considered as the reference process in the Double-strata Fuel Cycle Concept, another innovative concept is also being studied to improve the partitioning performance. This concept is called "ARTIST: Amide-based Radio-resources Treatment with Interim Storage of Transuranics" [3], where the object of the process is not the HLW but the spent fuel. The extract agents called BAMA and TODGA are used for uranium separation and all TRU separation, respectively, as shown in Figure 2. The advantages of the process are to use phosphorus-free agents consisting of carbon, hydrogen, oxygen, and nitrogen (CHON principle) to reduce the waste from the process. Basic study on the improvement of the process and the development of new extract agent is under way.

Figure 2. **Concept of ARTIST process**



R&D on transmutation fuel cycle in JAERI

JAERI has chosen the MA-nitride fuel as the first candidate for the dedicated transmutation system such as the ADS because of its possible mutual solubility among the actinide mononitrides and its good thermal properties. As the R&D for the fuel fabrication, high-purity nitrides such as NpN, AmN and (Cm,Pu)N were synthesised by the carbothermic reduction method and their material properties have been measured [4]. The solid solution of (Pu,Am,Cm)N was successfully prepared to demonstrate the mutual solubility. MA nitrides with inert matrix such as (Am,Zr)N, (Am,Y)N [5] and (Pu,Am,Cm,Zr)N were also synthesised successfully.

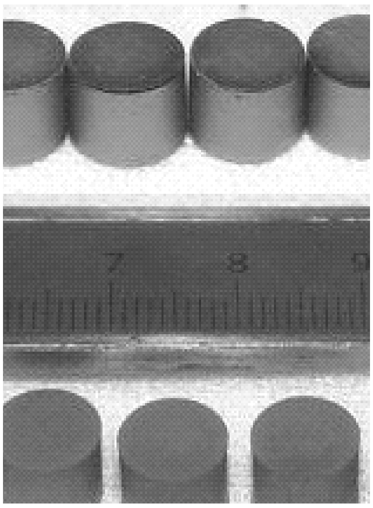
Chemical and thermochemical stability of AmN and (Am,Zr)N was studied in terms of the hydrolysis behaviour at room temperature and the evaporation behaviour at elevated temperatures. In both the cases AmN was stabilised by the formation of solid solution with ZrN. Thermal diffusivity, specific heat capacity and thermal expansion of Am-containing nitrides will be measured.

As for the irradiation test, (U,Pu)N fuel was irradiated in the experimental fast test reactor JOYO under the joint research with JNC, and no failure of fuel pins was found [6]. The irradiation test of uranium-free nitride fuels in the Japan Materials Testing Reactor (JMTR) was started in May 2002 [7]. One He-bonded fuel pin incorporating (Pu,Zr)N and PuN+TiN pellets (Figure 3), together with one (U,Pu)N fuel pin as reference, are under irradiation. The irradiation of the fuel pins will be continued till the end of this year, followed by post-irradiation examinations in 2005. Moreover, JAERI will participate in the FUTURIX programme to obtain the information of the irradiation behaviour of MA-bearing nitride fuels in PHENIX reactor in France.

Figure 3. **Appearance of uranium-free nitride fuel pellets**

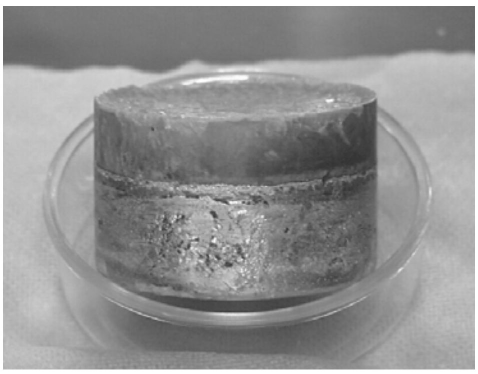
(Pu,Zr)N

PuN+TiN



To reprocess the irradiated MA nitride fuel, the pyrochemical process has been studied in JAERI because it has several advantages over the wet process in treating the dedicated fuel for transmutation including recycling of ^{15}N used in the nitride fuel. In the laboratory scale test, metallic Pu and Np were successfully recovered from non-irradiated PuN and NpN, respectively, by the molten-salt electrorefining technique [8]. Figure 4 shows the Cd cathode after recovery of Pu. A part of these R&D activities are collaborated with CRIEPI. Nitride formation behaviour of Pu-Cd alloy was experimentally investigated. Heating of Pu-Cd alloy at 973 K in N_2 stream resulted in almost complete nitride formation of Pu and distillation of Cd simultaneously.

Figure 4. **Recovery of Pu into liquid Cd cathode**



Regarding the transmutation target for long-lived fission products (LLFP), thermal properties of Tc-Ru were measured [9]. The basic experimental study is under way for the selection of suitable chemical form for the iodine target [10].

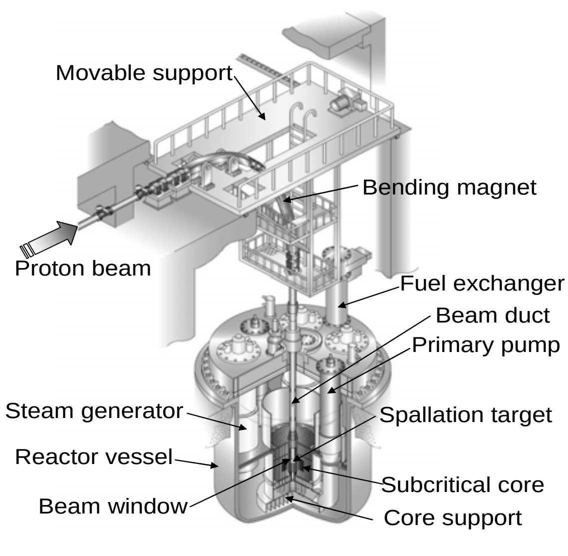
R&D on dedicated transmutation system in JAERI

The R&D of the ADS as the dedicated transmutation system is in progress for three technological areas: the accelerator, the lead-bismuth, and the subcritical reactor. Regarding the accelerator for the ADS, JAERI has chosen the superconducting LINAC as the first candidate. The cryomodule, which is a unit component of the high-energy part of the LINAC, is being fabricated to evaluate its acceleration performance, energy efficiency and stability [11]. In addition to this elemental development, JAERI is constructing a high-intensity proton accelerator under the framework of J-PARC Project (Japan Proton Accelerator Research Complex). The Phase-I of the project will be completed by the fiscal year (FY) of 2007, where a 200 MeV LINAC with 0.33 mA is being constructed as well as two synchrotrons. After the completion of the Phase-I, the LINAC will be upgraded to 400 MeV, which is the original specification of the Phase-I. After this upgrade, the LINAC will be extended up to 600 MeV by SC-LINAC as the Phase-II.

Regarding the lead-bismuth eutectic (LBE), three kinds of R&D activities are mainly under way in JAERI: the material corrosion/erosion test, the thermal-hydraulics test for the beam window and the evaporation test of polonium from LBE [12]. To perform these R&D, JAERI installed three LBE loops and a static corrosion test device. Moreover JAERI collaborates with the Mitsui Engineering and Shipbuilding Co. Ltd. for the corrosion loop test and the thermal-hydraulic loop test, and with JNC for the measurement of the evaporation rate of polonium. As the international collaboration for the LBE target demonstration, JAERI participates in the MEGAPIE Programme in PSI, Switzerland.

The design study of the subcritical reactor is also under way to propose feasible plant concept of 800 MWth ADS which can transmute 250 kg of MA annually [13]. The conceptual view of the subcritical reactor is shown in Figure 5. The LBE is adopted as the spallation target and the core coolant. The feasibility of the plant is being discussed in terms of structural strength of the beam window, the cooling performance of hot-spot pins, and so on [14]. The prediction accuracy of the reactor physics parameters such as the effective multiplication factor and the burn-up reactivity swing is also being discussed. To verify the accuracy of the MA nuclear data, the post irradiation examination (PIE) for the MA samples were implemented and the valuable information was obtained from its analysis [15].

Figure 5. **Conceptual view of 800 MWth LBE-cooled ADS**



As the next step to study the basic characteristics of the ADS and the transmutation technology, JAERI plans to build the Transmutation Experimental Facility (TEF) in the Phase-II of the J-PARC Project [16]. The construction of the TEF is scheduled to start around FY2007. The TEF consists of two buildings: the Transmutation Physics Experimental Facility (TEF-P) and the ADS Target Test Facility (TEF-T). The TEF-P is a zero-power critical facility where a low power proton beam is available to research the reactor physics and the controllability of the ADS. The TEF-T is a material irradiation facility which can accept a maximum 200kW-600 MeV proton beam into the spallation target of LBE.

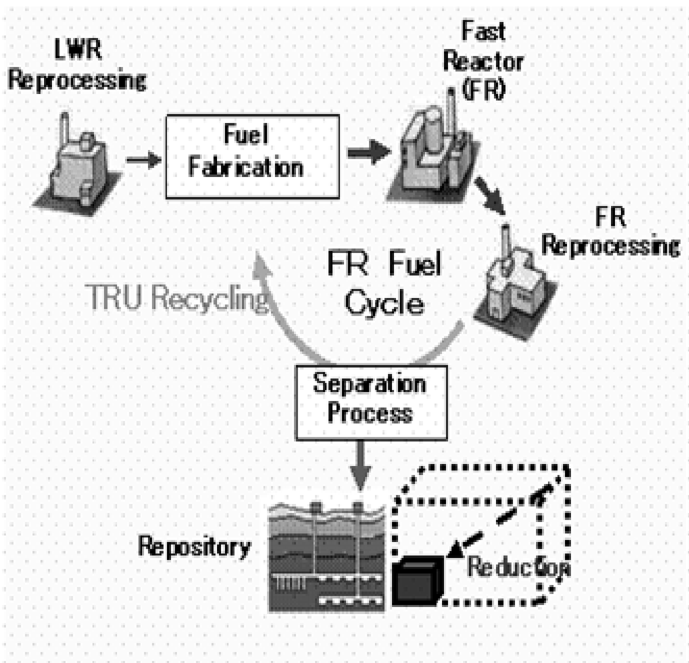
R&D on P&T waste disposal in JAERI

To evaluate the impact of P&T technology on the backend of the nuclear energy utilisation, the amounts and forms of the wastes from the partitioning plant and the transmutation fuel cycle were roughly estimated. The results showed that the volume of the vitrified forms can be reduced to about 1/4 by removing Sr-Cs, the platinum group and MA, and they can be disposed twice as densely as the normal case because of their low heat production. The Sr-Cs will be cooled as calcined waste forms, and its disposal method is being investigated. The amount of the wastes from the transmutation fuel cycle will be relatively small because the mass flow of the second stratum in the Double-strata Fuel Cycle Concept will be much smaller, approximately 1/100, than that of the first stratum [17].

Activities in JNC

Research and development of P&T in JNC have been performed in conjunction with the Feasibility Study on Commercialised Fast Reactor Cycle Systems (FS) [18] and as a part of it. Various candidate options, such as sodium-cooled/helium-cooled/lead bismuth-cooled/water-cooled reactors, MOX/metal/nitride fuels, aqueous/oxide electrowinning/metal electrorefining reprocessings, and palletising/sphere packing/casting fuel fabrications have been studied for the fast reactor fuel cycle system shown in Figure 6 under the FS.

Figure 6. **Advanced fast reactor fuel cycle**



The basic standpoints of JNC for the P&T are as following. All TRU will be treated in the same manner without distinguishing between Pu and MA, although necessity of cooling storage of Cm will be investigated. Homogeneous loading of TRU in fuel assembly will be mainly studied, although some extent of MA target fuel assembly will be studied. FP will be classified into four categories: the transmutation in the form of target assembly after separation (Tc, I), the cooling storage after separation (Sr, Cs), the waste as stable elements after separation (Mo, etc.), and the effective utilisation after separation (Ru, Rh, Pd, etc.).

Development goals of P&T technology are the reduction of radiotoxicity, the reduction of high level waste (HLW), and the effective utilisation of rare element FP.

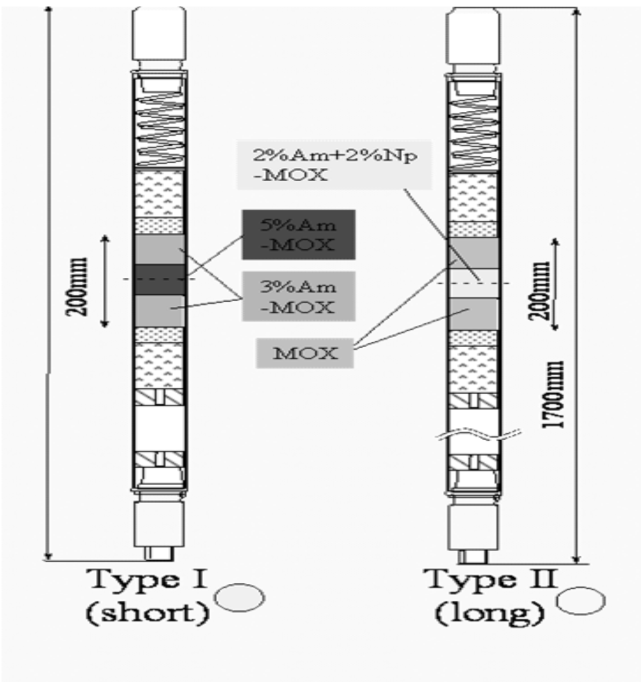
R&D on radiotoxicity reduction in JNC

High Recovery Rate Recycle of TRU

It has been evaluated that more than 99% recovery rate of TRU can be achieved and that there is a possibility to attain the target value of 99.9% by the NEXT (New Extraction System for TRU Recovery) process [19] which is an advanced aqueous reprocessing method developed by JNC. The NEXT process has also been evaluated to have an advantage in waste generation and cost for apparatus [20]. A basic research to develop the extraction system for total recovery of uranium and all transuranics from HLW solutions is in progress as a cooperative study with KRI (Khlopin Radium Institute), Russia [21]. The other hand, a cooperative research with Tokyo Institute of Technology to separate Am and Cm has started.

As for the MA contained MOX fuel development, fabrication experiments of 2%Np-2%Am-MOX pellet and up to 5% Am content MOX pellet have been successfully completed [22] and the irradiation experiment at the fast experimental reactor JOYO with the fuel pin structure shown in Figure 7 is scheduled in 2006.

Figure 7. **Fuel pin structure of MA-MOX irradiation experiment in JOYO**



Critical experiments of Np-loaded core which is a collaboration research with IPPE (Institute of Physics and Power Engineering), Russia, and irradiation experiment of MA (^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm) samples at JOYO have been conducted. Analysis works of both experiments are in progress in order to validate nuclear data and analysis method. It is suggested, through the analyses, that the value of one important parameter for MA composition change, ^{241}Am isometric ratio, would be around 0.85 [23].

P&T of LLFP

Candidate LLFPs to be transmuted in the FS are iodine and technetium. The most significant design limitation for LLFP sub-assembly is found, through the design study of LLFP loaded fast reactor core, to be the temperature of the moderator pins that determines the dissociated hydrogen permeation rate [24]. The out-of-pile tests in order to collect basic data such as sintering characteristics, compatibility with cladding and thermal conductivity have been started for candidate compounds of iodides such as NiI_2, MgI_2, BaI_2, CuI, KI, RbI, and YI_3. Sample irradiation experiment of iodides is scheduled from 2007 at JOYO.

As for the nuclear data, neutron capture cross sections for major FP such as ^{90}Sr, ^{99}Tc, ^{135}Cs, ^{137}Cs have been measured [25, 26] and a collaboration research with ORNL (Oak Ridge National Laboratory) to measure the neutron capture cross sections of ^{93}Zr, ^{99}Tc and ^{107}Pd is in progress [27].

R&D on HLW reduction in JNC

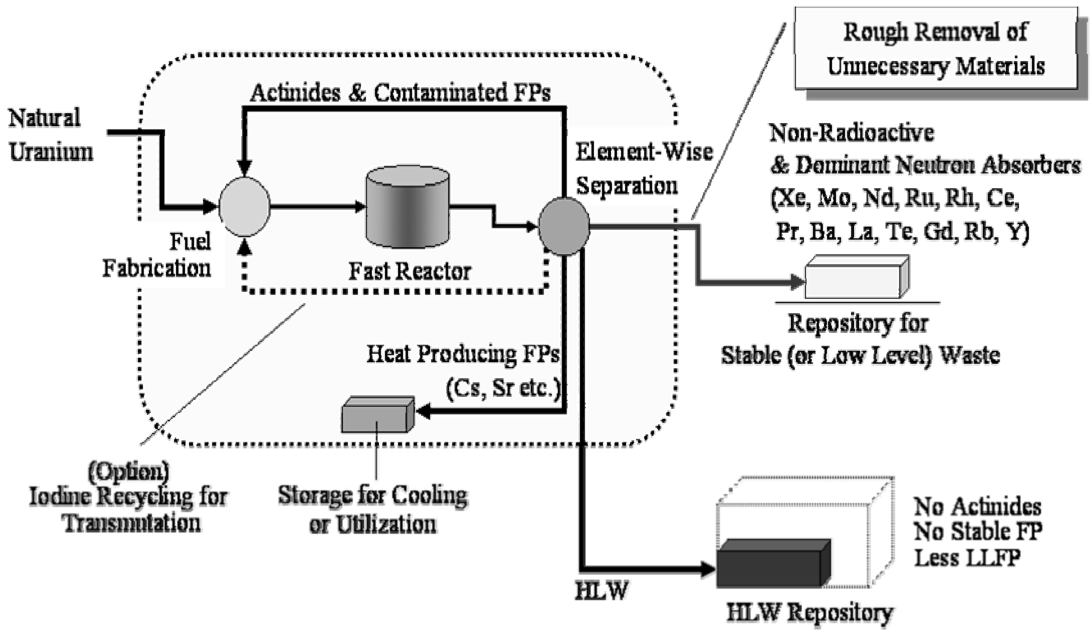
It has been evaluated by the FS that the generated wastes from the various candidate fast reactor fuel cycle options have no significant difference and amount to, in volume, about half of those from the conventional reference system which recycles without MA.

The advanced aqueous reprocessing method developed by JNC aims at to be a salt-free process. Consequently, the vitrified waste produced by the process can be reduced to about 60% of that of the conventional process with the aid of the fact that only small amount of heat generating TRU shifts, as recovery loss, to the HLW in the advanced fast reactor fuel cycle.

Reduction of vitrified waste can also be attained by removing heat generating FP, such as Sr and Cs, as well as Mo which causes soluble phase in glass. Therefore, design study of recovery and temporary storage methods of Cs/Sr and Mo including the economic evaluation have been conducted as a part of the FS.

JNC has proposed a new concept of fast reactor fuel cycle system named "ORIENT-cycle" (Optimisation by Removing Impedimental Elements) [28] which aims to minimise HLW by adopting an unconventional recycling scheme based on the idea "rough removal of unnecessary elements" instead of a conventional one "pure recovery of necessary elements" (see Figure 8). Stable FP that amount to about 60 wt% of all FP are identified as one of key "unnecessary elements". The evaluated result indicates that the ORIENT-cycle might be able to reduce the HLW from aqueous process by about one order of magnitude compared to the conventional process.

Figure 8. **Concept of ORIENT-Cycle**



Cooperative study with University of California, Berkeley, titled "Research on Effective Application of Partitioning and Transmutation Technologies to Geologic Disposal" was started from 2003 in order to evaluate quantitatively the reduction effect of geological repository burden due to the introduction of P&T technologies [29].

R&D on effective utilisation of rare element FP in JNC

An attempt to utilise relatively abundant rare metal fission products (RMFP) which amount to about 30 kg content per metric ton of fast reactor spent fuel due to its high burn-up is in progress. An electrolytic extraction method has been studied to separate RMFP (Ru, Rh, Pd, Tc, etc.) from the nuclear spent fuel to utilise them, for example, as catalysts [30]. A small scale experiment shown in Figure 9 indicates that the promising utilisation of RMFP will be as FP-catalyst for hydrogen production by water electrolysis.

Figure 9. **Experimental apparatus of electrolysis**

Basic research has started as another attempt of radiochemical approach to utilise Sr and Cs as radioactive source and heat source.

Activities in CRIEPI

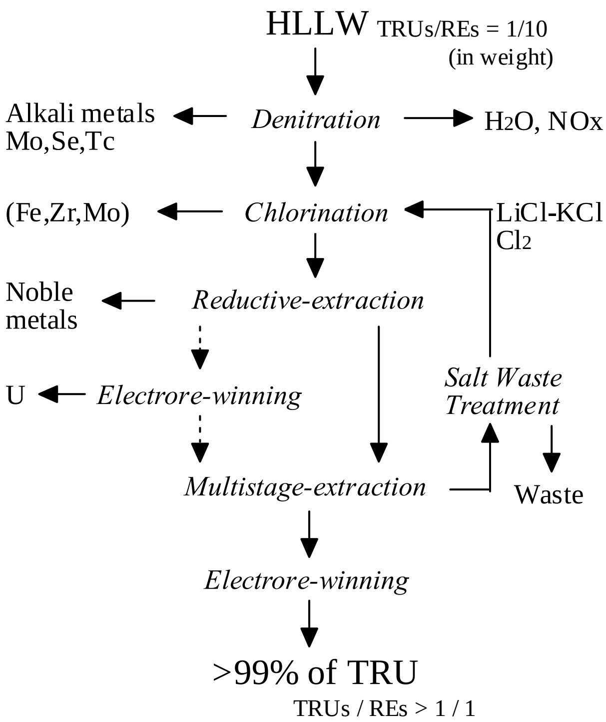
CRIEPI has proposed the P&T technology based on the metal fast reactor and its cycle, in which transuranium elements, TRUs, separated from high level liquid waste (HLLW) coming from PUREX reprocessing of spent LWR oxide fuels, should be transmuted [31]. The pyrometallurgical technique is used to separate of TRUs from HLLW. TRUs are mixed in alloy fuels of U-Pu-Zr for transmuting at fast reactors [32]. Spent metal fuels are reprocessed through pyrometallurgical process with electrorefining and reductive extraction, by which minor actinides, Np, Am, and Cm, are recovered together with U and Pu. This fact leads to an advantage on strong proliferation resistance as well as lightening an environmental burden at the waste disposal.

R&D on partitioning process in CRIEPI

Process Description

Figure 10 shows the process flow of separation of U and TRUs from HLLW in which rare earths are existing ca. ten times more than TRUs in weight. The HLLW has to be converted to oxide by heat-treating at 500^{o}C [33]. During this step Tc, Mo, Se and alkali metals are expecting to be separated. The oxides are changed to chlorides by reacting with chlorine gas at 700^{o}C in LiCl-KCl salt bath, in which Fe-, Zr-, and Mo-Clx with high evaporation rate are captured into molten chloride bath trap. The first step of reductive extraction in a system of LiCl-KCl/Cd is applied to separate noble metal elements, and then multistage extraction in a same salt and liquid metal, Cd or Bi, system is possible to separate U and TRUs from rare earths with separation efficiency of more than 99% with attaining the decontamination factor of over 10. Both reduction steps require lithium metal as reductant, and are operated at 500°C. The advantage of this process is that most of solvents, molten salt, liquid metal and chlorine gas can be recycled after treatment [34].

Figure 10. **Process flow of pyro-partitioning**



Previous and current activity

In the system of molten chloride and liquid metal, the separation of actinides from lanthanides is a key issue from thermodynamic aspect. In CRIEPI, the electrochemical potentials were measured in a cell of M/MCl_n-LiCl-KCl//AgCl-LiCl-KCl/Ag as a function of molar fraction of MCl_n in LiCl-KCl. M denotes an actinide or lanthanide element. Applying Nernst equation on electrochemical potentials measured gives the standard potentials of actinides and lanthanides [35-38]. The potentials promise the precise prediction for separation.

In order to examine the separation efficiency between actinides and lanthanides by reductive extraction, the distribution coefficient of each element was measured in systems of LiCl-KCl/Cd and LiCl-KCl/Bi at 450-500°C, from which the separation factor between actinide and lanthanide is obtained [39, 40]. The LiCl-KCl/Bi system has a higher potential than LiCl-KCl/Cd system for the separation between both elemental groups.

Following the measurements of electrochemical potentials and distribution coefficients, separation tests of actinides have been carried out by reductive extraction. It is assured by multistage reductive extraction experiments with simulated waste that more than 99% of each actinide can be recovered [41]. Through these experiences, it is convinced that the verification with genuine hot material makes an important role for finalising process flow.

The caisson shown in Figure 11 and Figure 12 has been prepared in hot cell facility by the cooperation of the Institute of Transuranium Elements to use of genuine HLLW produced from reprocessing of irradiated MOX fuels. Prior to the experiment using genuine material, the electrorefining study has been performed to recover Pu from metal fuel for fast reactor fuel cycle programme in CRIEPI. Figure 13 shows the U-Pu-Zr alloy fuel before and after experiment [42].

Figure 11. **External view of the caisson**

Figure 12. **Internal view of the caisson (Electrorefiner operated with manipulator)**

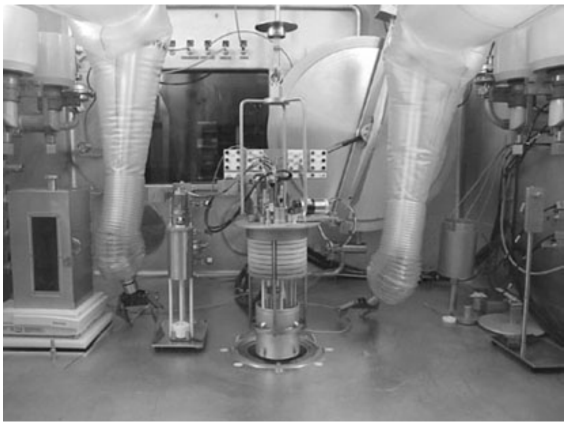
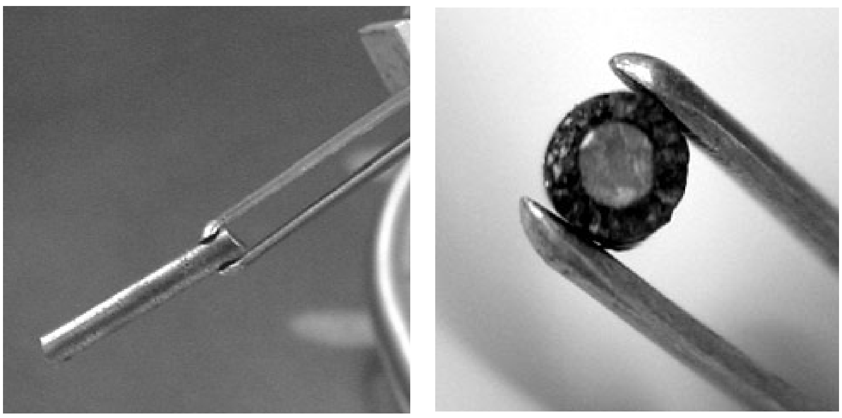


Figure 13. **U-Pu-Zr alloy (left) before and (right) after experiment. U-Pu-Zr alloy remaining after experiment was surrounded by dense layer with Zr metal and salt mixture**



Currently, the programme proceeds to use the genuine HLLW. Some experiments are carrying out to convert HLLW to oxides by evaporating water and decomposing nitrate solution. Chloride conversion and TRUs separation by multistage reductive extraction by use of this oxide are planned in this Japanese FY.

R&D on transmutation in CRIEPI

Characterisation of alloy with minor actinides

The actinides are recycled into metal fuel of fast breeder reactors. Phase diagram predicts a low limited solubility of tri-valence species in U-Pu-Zr matrix [43]. U-Pu-Zr alloys added with minor actinides of 2 wt% and 5 wt% each and lanthanides of 2 wt% and 5 wt% each were casted in order to evaluate the miscibility between tri-valence species and U-Pu-Zr. Metallography shows the intermetallic compound of americium, neodymium and cerium distributing uniformly at grain boundaries [44]. The solubility of tri-valence species in U-Pu-Zr is less than 1 wt% even at melting state as observed in EPMA analysis.

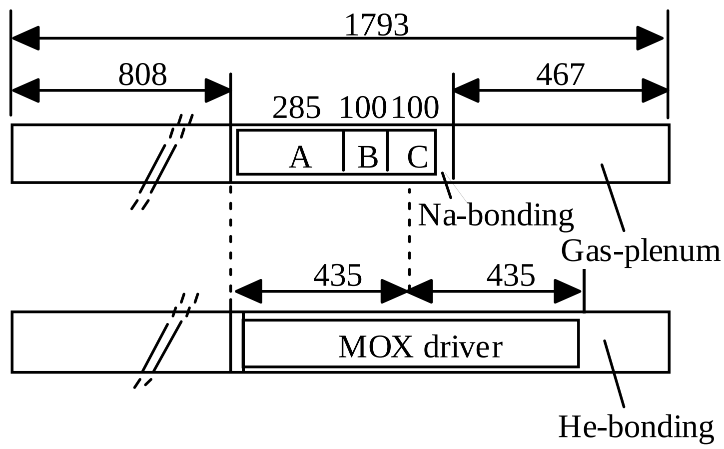
For use of the metal alloy as nuclear fuel, modification of the fundamental properties, such as melting temperature, elastic modulus and redistribution of components under a temperature gradient, has to be clarified on alloys of U-Pu-Zr added with minor actinides and lanthanides [45]. It was observed that the properties of the alloys added with minor actinides and lanthanides up to 5wt% each are approximately same with those of U-Pu-Zr.

The liquefaction temperature by eutectic formation of U-Pu-Zr fuel and stainless steel cladding is expected to be above 600^{o}C, which affects directly the reactor operation temperature. The liquefaction was evaluated from diffusion tests using U-Pu-Zr/Fe couple and the phase diagrams were assessed both by thermodynamic calculation and metallography. The diffusion tests indicate that no melting phase appears under 650^{o}C in a region with less than 25 wt% of plutonium in fuel U-Pu [46, 47]. The thermodynamic calculation of the activity gradient explains the mechanism of liquefaction, namely the diffusion paths through a two-phase region of liquid and U_6Fe, provided that a cladding temperature is greater than 650^{o}C.

Irradiation study of minor actinide containing Alloys

The irradiation study for minor actinide-containing alloy, i.e. U-Pu-Zr-MA(Np, Am, Cm)-RE(Nd, Ce, Gd, Y), is in progress. Nine pins with various concentrations of MA and RE have been fabricated by casting and power metallurgy for high amount of MA and RE. Figure 14 shows figure configuration of irradiation pins for Phenix Fast Reactor. Instead of MOX driver pins, sodium-bonded metal fuel pins shown in the figure were prepared at the Institute of Transuranium Elements. Three kinds of pins, in which 2%MA-2%RE, 5%MA-5%RE and 5%MA in U-Pu-Zr and U-Pu-Zr for comparison are sandwiched with U-Pu-Zr standard alloy, are allocated in a subassembly. Three assemblies with same configuration of pins, METAPHIX -1, -2, -3, are loaded in a core region of PHENIX at the end of 2003, and start the irradiation at the rated output from January 5, 2004. Each assembly will be irradiated to low, medium and high burn-up. METAPHIX 1 discharged from the reactor on August 5, 2004 after a burn up of 2,4 at%. It is also planned that METAPHIX 2 will be unloaded beginning of 2006 with 7 at% and METAPHIX 3 in 2008 with 11 at %. After unloading and cooling, non-destructive examination will be planned for integrity investigation. After NDE, pins will be transported to the Institute for Transuranium Elements, where the destructive examinations, transmutation rate of MA, physical and chemical behaviour of MA in alloys are expected.

Figure 14. **Irradiation test pins for METAPHIX (unit : mm) A, C: U-Pu-Zr reference specimen; B: U-Pu-Zr-MA-RE specimen (diameter of pin = 6.55 mm, cladding = 15/15Ti cw)**



Concluding remarks

The R&D activities for P&T technologies by three Japanese institutes were summarised in this report. JAERI is engaging in the Double-strata Fuel Cycle concept consisting of the partitioning process of the high-level waste and the dedicated transmutation cycle using the accelerator driven system (ADS) fuelled with the minor actinide (MA) nitride fuel. JNC and CRIEPI are engaging in the R&D on the P&T technology using commercialised fast reactors (FR), where JNC is mainly in charge of the MOX fuel and the aqueous reprocessing, and CRIEPI is mainly in charge of the metallic fuel and the dry reprocessing. It should be noted that the unification of JAERI and JNC is scheduled in 2005 so that the major part of the R&D in this report will be continued in the new integrated institute.

Furthermore, in Japan, many activities relating to P&T technologies are also implemented by universities. The Tokyo Institute of Technology (TITech) started a basic study for innovative separation/transmutation systems toward vanishing high-level wastes under a framework of "21st Century Center of Excellence (COE) Programme on Innovative Nuclear Energy Systems for Sustainable Development of the World (COE-INES)". TITech is also conducting the measurement and the evaluation of nuclear data for the MA transmutation by collaborating with JAERI, JNC, Kyoto University, and so on. Kyoto University and the High Energy Accelerator Research Organisation (KEK) started the construction of a new-type proton accelerator to couple with an existing critical assembly, though the purpose of this programme is not only for the study of the transmutation but also for the energy production. The Atomic Energy Society of Japan has established the Research Committee on Partitioning and Transmutation Cycle to promote and integrate the various activities mentioned above and to discuss the effect of P&T technology in the context of the waste management.

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OVERVIEW OF THE UNITED STATES P&T PROGRAMME

Carter Savage
AFCI Programme Director, US Department of Energy

The National Energy Policy and nuclear power

"The NEPD Group recommends that the President support the expansion of nuclear energy in the United States as a major component of our national energy policy."

Report of the National Energy Policy Development Group, May 2001

Recommendations:

- Support expansion of nuclear energy in the United States.
- Develop advanced nuclear fuel cycles and next generation technologies.
- Develop advanced reprocessing and fuel treatment technologies.

The United States Department of Energy has a number of initiatives to promote the growth of nuclear energy:

Nuclear power 2010

- Explore new sites.
- Develop business case.
- Develop Generation III+ technologies.
- Demonstrate new licensing process.

Advanced fuel cycle initiative

- Recovery of energy value from SNF.
- Reduce the inventory of civilian Pu.
- Reduce the toxicity & heat of waste.
- More effective use of the repository.

Nuclear hydrogen initiative

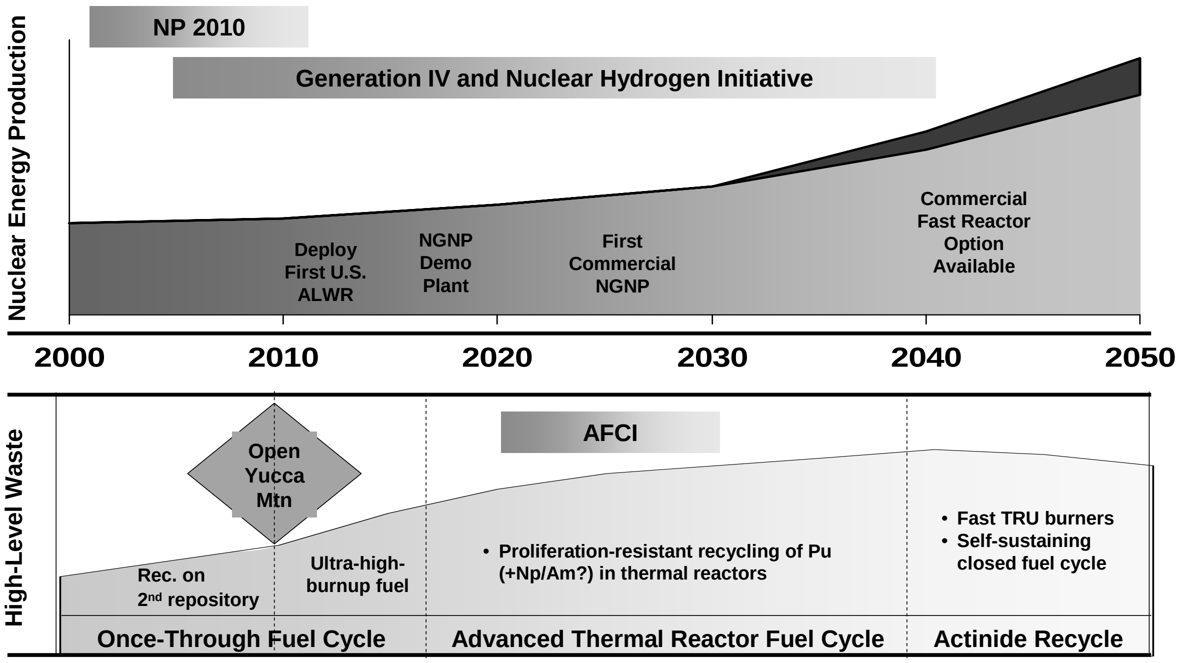
- Develop technologies for economic, commercial-scale generation of hydrogen.

Generation IV

Better, safer, more economic nuclear power plants with improvements in

- safety and reliability;
- proliferation resistance and physical protection;
- economic competitiveness;
- sustainability.

A Long-term U.S. strategy for nuclear energy



U.S. Generation IV implementation

Generation IV top priority – Next generation Nuclear Plant

- Collaborative with international community.
- Collaborative with industry, especially utilities.
- Demonstrate H_2 and direct-cycle electricity production.
- Result in a commercially viable plant design.

Generation IV second priority

- GFR
- LFR
- SFR

U.S. Fast Reactor

Closely coordinated with

Advanced Fuel Cycle Initiative

Lower priority

- SCWR
- MSR

Advanced fuel cycle initiative (‘AFCI’)

Mission

- Develop proliferation-resistant spent nuclear fuel treatment, fuel and transmutation technologies to enable the transition from the once-through fuel cycle to a stable, long-term, environmentally, economically, and politically acceptable advanced closed fuel cycle.

Goals

- Develop advanced fuel and fuel cycle technologies for application to current operating commercial reactors and next-generation reactors.

- Develop technologies to reduce the cost of geologic disposal of high level waste from spent fuel, enhancing repository performance.

ACFI Benefits

Achieving AFCI programme goals could:

- Reduce civil plutonium inventories, reducing proliferation risk.
- Extract valuable energy from spent fuel components.
- Retain nuclear energy as a major component of the U.S. energy mix, ensuring energy security in the 21st century.
- Significantly reduce volume, heat load and radiotoxicity of high-level waste from spent fuel, delaying any near-term need for a second geologic repository in the U.S.

History of Department of Energy’s Advanced Fuel Cycle Research

- **1999 – Accelerator Transmutation of Waste (ATW) –** roadmap issued by RW, outlined use of high-powered proton accelerators for destruction of all actinides from spent fuel.

- **2000 – ATW –** research programme initiated to explore transmutation technology ($9M).
- **2001 – Advanced Accelerator Applications (AAA) programme launched –** combined ATW with Accelerator Production of Tritium (APT) programme to optimise use of resources ($34M-NE, $34M-DP).
- **2002 – AAA refocused to AFCI –** emphasis on reactor based systems, accelerator transmutation focused on "fuel burn" role to minimise toxicity and support Generation IV (Gen IV) fuel development ($50M).
- **2003 – AFCI establishes new management structure –** National Technical Directors, Technical Integrator, and integrates with Gen IV for fuel cycle development ($58.2M).
- **2004 – AFCI Budget – $68M.**
- **2005 – AFCI Budget Request – $46.3M.**

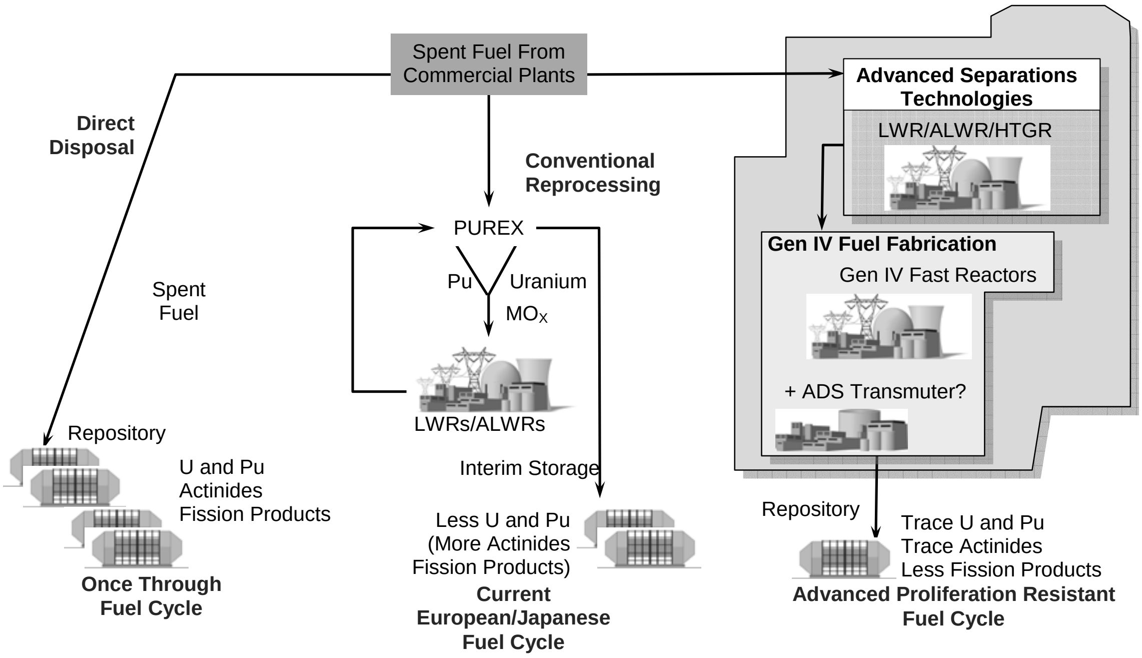
ACFI recent accomplishments

- Successfully demonstrated UREX aqueous process at SRNL, separating uranium from actual spent nuclear fuel with over 99.99 percent purity.
- Demonstrated lab-scale high-purity separation of cesium/strontium, plutonium/neptunium and americium/curium from spent fuel (INEEL, ANL, ORNL).
- Fabricated and irradiated non-fertile and low-fertile metallic, nitride and oxide fuel samples containing plutonium, neptunium and americium (LANL, ANL-W, INEEL). PIE started at ANL-W.
- Built a lead-bismuth test loop at LANL and completed 1000 hour corrosion test.

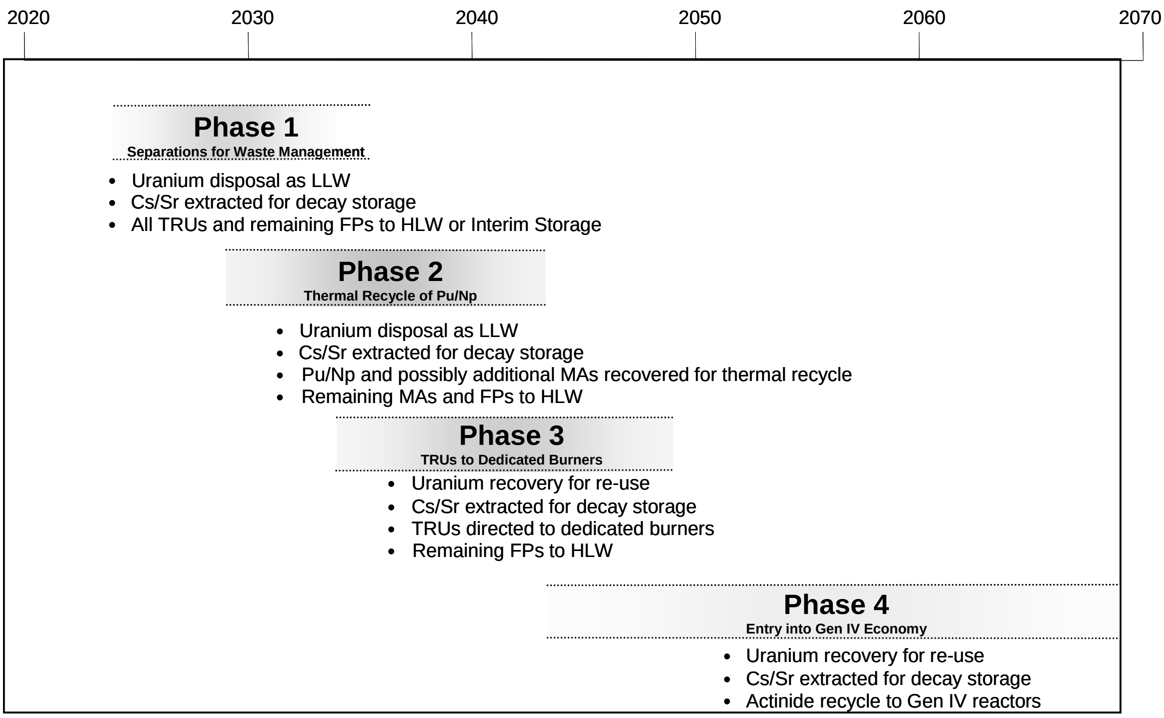
AFCI challenges

- Define, plan and execute the optimal research to inform the 2007-2010 Secretarial recommendation on second repository and meet AFCI/Gen IV programme goals.
- Scale of demonstrations to provide sufficient confidence in 2007-2010 Secretarial recommendation.
- Integration of analysis and modelling with experiments.
- U.S. non-proliferation policy.
- R&D facilities, including pilot-scale demos and fast spectrum irradiation facility.
- Reliable long-term funding.

AFCI approach to spent fuel management



AFCI long-range strategy



Separations – Current approach

- Aqueous separations process development (UREX+1, UREX+2) with laboratory-scale experiments.
- Process technology development (equipment, process integration, process control and instrumentation, safeguards instrumentation, etc.).
- Development of waste forms and storage forms (including performance testing).
- Evaluation of advanced processing methods and validation of promising candidates at laboratory-scale.

Separations technology development in 2005

- Large centrifugal contactor tests.
  - Scale-up issues, remote operation/reliability/maintainability.
  - Process sampling and analysis, process control.
- Dissolution studies.
  - Optimise for most complete dissolution of TRU and compatibility with subsequent separations steps.
- Feed clarification experiments.
  - Efficiency of different methods.
- Alternative head-end process development.
  - Voloxidation process.
  - Off-gas recovery and treatment.
- Uranium crystallisation process development.
  - Maximising purity of separated uranium.
  - Carbonate dissolution process.

Advanced fuels research

- NGNP particle fuel.
  - UCO, SiC coating.
  - High temperature requirement (1 000°C).
- LWR Recycle Fuel.
  - Mixed Oxide.
  - Pu + Np + Am? + Cm?

  - Inert Matrix.
  - Intrinsic proliferation resistance.

- Fast reactor Fuels.

  - Metal, nitride, oxide, dispersion.
  - Optimise transmutation.

Materials research

- Coolants/targets for Generation IV fast reactors and Accelerator Driven Systems.

  - Lead, lead-bismuth for LFR, ADS.
  - Helium, supercritical CO2 for GFR.

- Structural materials for high-temperature, high fast neutron flux performance.

- Fuel matrix materials for very high-burnup fast reactor and transmutation fuels.

AFCI international collaborations

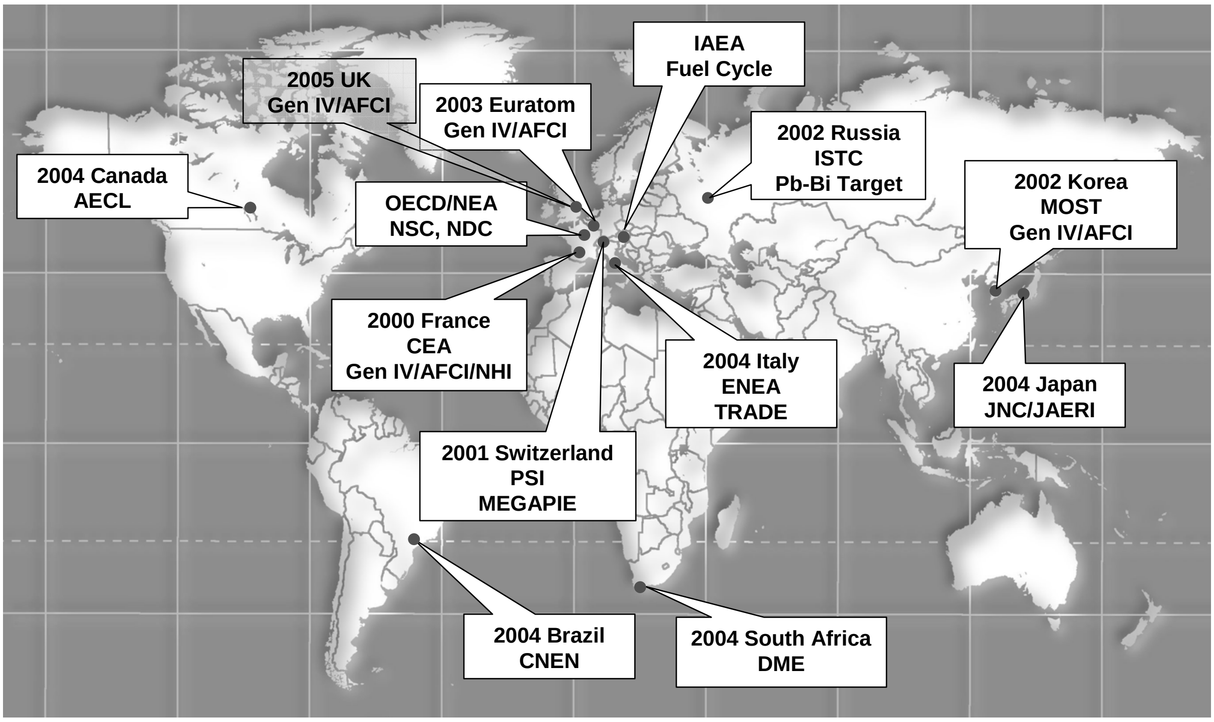
- International Cooperation has provided U.S. with much needed research and experimental data.
- France – CEA: separations, fuels (FUTURIX), physics, systems studies.
- MEGAPIE facility at Paul Sherrer Institute (Switzerland); spallation target technology, physics & engineering support
- Russia – LBE Test target; UNLV cooperation
- OECD/Nuclear Energy Agency
- European Commission
- Japan
- South Korea
- IAEA

Department of Energy approach for international collaborations

- International Nuclear Energy Research Initiative (INERI) changes in FY 2004.
- INERI budget funds completion of ongoing projects only; no new starts.
- New starts of bilateral international collaborations funded by the research programmes (AFCI, Gen IV, Hydrogen).
- INERI bilateral agreements will be main mechanism (France-CEA, S. Korea, OECD/NEA, Euratom, Brazil, Canada); several new agreements close to signing (Japan, South Africa, UK).

- Existing AFCI cooperative agreements and “implementing arrangements” will also be used.
- Collaborations with European community on FUTURIX, MEGAPIE, TRADE expected to continue.
- Trilateral with France and Japan under discussion for use of Monju for transmutation fuel assembly tests.

International collaborations



FRENCH WASTE MANAGEMENT STRATEGY FOR A SUSTAINABLE DEVELOPMENT OF NUCLEAR ENERGY

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Abstract

The production of nuclear energy in France has been associated, since its inception, with the optimisation of radioactive waste management, comprising the partitioning and the recycling of recoverable energetic materials, volume reduction, the conditioning, and the disposal of the end waste.

The public's concern regarding its long-term management led the French Government in 1990 to delay the implementation of the geological disposal and to prepare a law passed on December 30, 1991, requesting in particular the study of solutions and processes for:

- minimising the quantity and the hazardousness of waste, via partitioning and transmutation;
- disposing the waste either reversibly or irreversibly in deep geological formations;
- waste conditioning and long-term interim storage.

The law stipulates that, by 2006, the research results obtained should be documented in the form of a general report to be submitted to members of parliament. The law also created the National Evaluation Commission to supervise the research work on an annual basis.

At the time of the implementation of measures comprised in the Law, the Government commissioned CEA to conduct research in the area of Line 1 (partitioning and transmutation) and Line 3 (conditioning and interim storage) and Andra on Line 2 (geological disposal).

The feasibility of partitioning did not appear easily accessible at the time the research began. Its feasibility was demonstrated in 2001 following a series of tests conducted on actual solutions of dissolved spent fuel, in the CEA's Atalante installation at Marcoule. The 2002-2005 programme encompasses technological demonstrations, with representative process equipment, and an economic evaluation of the industrial implementation of partitioning.

Studies on transmutation, which were initiated before the Law of 1991, rapidly led to concluding that transmutation of minor actinides (Americium, Curium, and Neptunium) was feasible in particular in fast neutron systems.

The work on transmutation is now focusing on a demonstration of technology feasibility and this is considered in relation with the development of future nuclear energy systems capable of minimising their production of long-lived radioactive waste, within the framework of the Generation IV International Forum.

For supporting the objective of a sustainable nuclear energy, such Generation IV systems are envisioned with fast neutrons and a global recycling of all actinides from the spent fuel to be capable of burning fissile materials, breeding fertile materials and transmuting minor actinides. Furthermore, a group management of actinides is aimed at achieving an appropriate resistance to risks of proliferation.

Interim storage is a mode of package management ensuring, by design, the protection of waste package and their recovery at a later date, under safe and technically established conditions. It is a factor of flexibility for all considered waste management strategies, in the medium or the longer term. The work done in the context of Line 3 of the 1991 Law has comprised the identification and the development of long-term interim storage concepts. The 2003-2005 programmes are targeted at the technological development of canisters and specific interim storage components, using demonstrators and qualification programmes.

Important results are now available, on the one hand concerning the possibility of significantly reducing the quantity and the radio-toxicity of long-lived waste, and also relating to the modes of waste conditioning and long-term interim storage facilities.

With these results combined with those obtained by the Andra on geological disposal, technical solutions that can be implemented progressively, will be presented for decision in 2006 to manage the high level and long-lived radioactive waste generated by the LWRs. Most of these solutions are seeds for the advanced fuel cycle processes needed to manage the group actinides streams and the waste forms generated by vision of sustainable nuclear energy that Generation IV systems constitute. These scientific achievements concerning waste management must be accompanied by a democratic and political debate at the French parliament.

Introduction

The production of nuclear energy in France has been associated, since its inception, with the optimisation of radioactive waste management, comprising the partitioning and the recycling of recoverable energetic materials, volume reduction, the conditioning, and the disposal of the end waste. [1]

Low-activity and short-lived waste (approximately 15 000 m^3/year) mainly originates from the operation of nuclear generating plants, is managed by the Andra (the French National Agency for Radioactive Waste Management) and disposed on surface at the Centre de l'Aube (and formerly at the Centre de La Manche, closed down since 1997).

The actual industrial reprocessing operations remove and recycle the plutonium, which is a highly energetic material and also the primary contributor to long-term radiotoxicity, and condition the long-lived waste in a safe and durable manner, and in a very small volume (130 m^3/year for HLW glasses and 320 m^3/year for medium-activity wastes).

Nevertheless the decay period of the waste remains long, and the public's concern regarding the long-term management of high level and intermediate level waste made the French Government in 1990 to delay the implementation of the geological disposal and to prepare a law, passed on December 30, 1991, requesting in particular the study of solutions and processes for:

- minimising the quantity and the hazardousness of waste, via partitioning and transmutation;
- either reversibly or irreversibly disposing the waste in deep geological formations;
- waste conditioning and long-term interim storage.

At the time of the implementation of measures comprised in the Law, the Government commissioned CEA to conduct research in the area of Line 1 (partitioning and transmutation) and Line 3 (conditioning and interim storage) and Andra on Line 2 (geological disposal). We will present here an update on the progress made by the research. Research has been conducted in a very sustained way since 1992 and benefits from major co-operations in France (EDF, AREVA, Andra, CNRS, universities…), Europe, and internationally.

For line 1: partitioning and transmutation, the actual industrial reprocessing operations remove and recycle the plutonium, so the reprocessing greatly reduces both the waste radiotoxicity and its lifetime (from a few hundred thousand years in an open cycle to a few tens of thousand years in a closed cycle). In the most advanced fuel cycles using fast-spectrum reactors and extensive recycling, it may be possible to reduce the radiotoxicity of all wastes such that the isolation requirements can be reduced by several orders of magnitude (e.g., for a time as low as 1 000 years) after discharge from the reactor. This would have a beneficial impact on the design of future repositories and disposal facilities worldwide.

Line 1 researches concerns future wastes which could be produced by enhanced reprocessing mainly in the perspective of sustainable development of nuclear energy.

1. France produces, each year and per capita, approximately 1 kg of waste originating from the production of nuclear energy, compared to 2 500 kg of industrial waste, which includes 100 kg of toxic chemical waste.

The management of nuclear waste and the access to cost effective natural uranium resources will assuredly be major challenges for nuclear energy systems over the 21st century, taking into account the rapid growth of primary energy demand and the increased share of nuclear power to meet both needs of electricity generation and hydrogen production for the transports, while limiting greenhouse gas emissions.

Legacy wastes and wastes which are actually produced in France by reprocessing are ultimate wastes non relevant of line 1 operations, in fact researches have shown that nuclides retrieval from actual conditioned wastes would be technically very difficult and non interesting from wastes volume and economics points of view.

These wastes are concerned by studies relevant of line 2: Disposal (Andra) and line 3: Conditioning and interim storage.

Legacy and actual wastes

The R&D work in line 3 comprises two chapters:

- conditioning, which entails the making of and the knowledge base about the waste package: development of radioactive material conditioning processes, canisters, package characterisation and long-term behaviour studies;

- interim storage, focusing on the definition and the qualification of long-term interim storage concept designs, on surface or subsurface.

Conditioning

Treatment and conditioning processes

Developments made in the area of waste treatment and conditioning were targeted at ensuring the availability of qualified processes that could be applied to historic waste to be recovered or to improve (volume reduction) a number of existing treatment processes: waste vitrification in cold-crucible, bituminisation of treatment effluent residues currently stored.

Overall, these developments have reached their objectives and are, for the most part, complete.

Canisters

A major effort is being made in the area of development and qualification of canisters for long-term interim storage or disposal. The canister is an external envelope around the internal primary package, providing an additional barrier (useful for handling, recovery and reopening [reversibility], tightness, durability, etc.).

Developments include in particular:

- Canisters for dry interim storage of spent fuel: [2] in order to guarantee the next recovery step under safe conditions, each fuel assembly is placed within an individual stainless-steel case kept under inert (helium) atmosphere; cases are then placed within a canister developed for interim storage. The technological canister demonstrators will be available at Marcoule in 2004. CEA, EDF, and Andra are also developing, following similar principles, a disposal canister compatible with interim storage. The demonstrator will also be available at Marcoule in 2004 for qualification.

- Canisters for interim storage of medium-activity long-lived waste:

  - Concrete canisters with a holding capacity of one to several primary packages, these canisters conceived with Andra are common for storage and disposal;

  - Primary canisters ensuring physico-chemical compatibility between the waste and the surrounding canister.

The technological demonstrators will be available at Marcoule in 2004.

Package long-term behaviour studies

The objective of the conditioning being to ensure durable confinement for all the steps in the package management, it is necessary to establish scientific and technical ground for the prediction of the package long-term behaviour, and to confirm that the relevant functions are provided, in particular confinement, handling, and recovery.

The work done in recent years has provided the scientific base of a true science on long-term behaviour, with detailed modeling and experimental validation of the principal phenomena at work considering all the different types of packages (glass, concrete, bituminised, compacted waste, spent fuel), and of the evaluation of their performance with regard to the durability of confinement in interim storage (physico-chemical evolution of the materials and prediction of their state at the time of recovery and handling operations) or in disposal (in particular under typical conditions of alteration by water in the very long term). The long durability of glass, in particular, has been established.

All the phenomena affecting long-term behaviour for each type of packages are integrated into qualified "operational models" which are used in global studies looking into the long term waste management facilities operation and behaviour over time (interim storage or disposal). These programmes benefit from the expertise of international experts in the fields.

2. The French downstream strategy does not consider spent fuel to be a waste form, since this would be contradictory to the goal of minimising its long-term hazardousness; nevertheless, in order to explore all the various options, the research conducted under Line 3 on conditioning and interim storage considers long-lived radioactive waste (vitrified high-activity long-lived waste, all medium-activity waste), and also spent fuel, as a whole. In 1992, the CEA undertook, in cooperation with EDF, COGEMA and Andra, a study of conditioning, interim storage, and disposal of spent fuel, which resulted in a report in 1996.

Package characterisation and inspection

Research conducted has resulted in the development of very thorough methods and systems of package characterisation and inspection, via non-destructive measurements or via samplings and radio-chemical analyses, in order to confirm that any package can meet the acceptance criteria developed for the interim storage or disposal facilities.

Interim storage

Interim storage is a mode of package management ensuring, by design, the protection of waste package and their recovery at a later date, under safe and technically established conditions.

By design, the interim storages can be constructed and operated to accommodate the long term ("*aptitude séculaire*") interim storage of waste package. The feasibility of such interim storage facilities is demonstrated, and site characteristics being used are sufficiently broad for conducting the research in a generic way.

Preliminary design studies of two facility concepts, on surface and subsurface, will be achieved at the end of 2004.

Geological Disposal

Researches conduct by Andra must enable the evaluation of disposal feasibility in clay. These programmes include study and presentation of a solution for geological disposal with scientific questions which have been addressed.

The feasibility evaluation is based upon a characterisation of geological medium with global models founded on results obtained in Bure underground laboratory. Main questions which will be addressed are absence of fracturation, comprehension of the behaviour of altered zone and confirmation of favorable hydrological and geochemical properties. A drilling programme achieved in 2004 brings data on these points in order to demonstrate the geological disposal feasibility.

Future Wastes

Studies on partitioning and transmutation aim at isolating the most radiotoxic long-lived elements present in the waste (primarily minor actinides, which are vitrified with the fission products), then at transmuting them through recycling in nuclear reactors, in order to change them into non-radioactive or shorter-lived elements.

Commercial processing-recycling of plutonium is the first necessary step, the partitioning of minor actinides (americium, curium and neptunium), followed by their transmutation, would reduce to a few hundred years the time necessary for the radiotoxicity of the vitrified waste to become similar to that contained in the natural uranium ore originally used.

Such "light glass" packages will be the future ultimate waste form destined to the geological repository.

Partitioning

The on-going programme since 1992 comprises two chapters: *PURETEX* (which was completed in 1998) and *ACTINEX* (partitioning).

The PURETEX programme goal was the minimisation of the waste stream originating from the spent-fuel processing operations in the UP3 plant at La Hague, commissioned in 1989, the aim being to reduce the volume of medium-activity waste and diminish the activity released in liquid and gaseous effluents.

Overall, results obtained entail a threefold reduction in the volume of solid waste and a tenfold reduction in the activity of liquid effluents since the start of the reprocessing plant.

The *ACTINEX* programme deals specifically with the partitioning of the long-lived radionuclides contained in the waste.

The feasibility of partitioning did not appear easily accessible at the time the research began because chemical species to separate have very similar properties. Therefore a new chemistry for partitioning elements had to be developed. One hundred very selective new molecules have been tried for minor actinides (the feasibility of separation of neptunium, iodine and technetium has been established; it is based on adjustments of the reprocessing process). Three molecules (two molecules of the family of the diamides and one organic acid) were identified to be used in the partitioning process for americium and curium. Its feasibility was demonstrated in 2001 following a series of tests conducted on actual solutions of dissolved spent fuel, in the CEA's Atalante installation at Marcoule. The process partitioning performances are very satisfactory (~ 99.9% of the minor actinides recovered). Specific molecules (of the calixarene family) for cesium extraction have also been developed, and the feasibility of its partitioning has also been established, with similar performances.

In 2005, the programme encompasses technological demonstration with representative process equipment, that is to say real testing of these processes involving fifteen kilogrammes of spent fuel in hot cell in Atalante at Marcoule, and economic evaluation of industrial implementation of partitioning.

Specific conditioning processes for the partitioned elements which could wait for transmutation are also being developed (reversible conditioning for interim storage).

Transmutation

Studies on transmutation, which were initiated before the 1991 Law, rapidly led to concluding that transmutation of minor actinides (americium, curium, and neptunium) was feasible in particular in fast neutron spectra. This is linked to the thorough knowledge of transmutation yields resulting from developments in reactor physics and from qualification in this field. However, the transmutation of long-lived fission products (such as iodine 129, cesium 135 and technetium 99) appears less attractive (marginal reduction of radiotoxicity compared to that achieved by transmutation of minor actinides) and not very promising (low transmutation yields).

Results obtained show that the feasibility of transmutation is demonstrated, in pressurised-water reactors (recycling and transmutation of plutonium), in advanced systems of nuclear-energy production (fourth-generation fast-spectrum reactors, with recycling and transmutation of all heavy nuclides, uranium, plutonium, the minor actinides) and in dedicated subcritical incinerator reactors. The best conditions for transmutation are obtained in fast neutron spectra.

Work on transmutation is now focusing on technical elements necessary for the demonstration of its technological feasibility.

Conceptual studies of subcritical systems dedicated to transmutation (ADS Accelerator Driven Systems consisting of a nuclear reactor operating subcritically and fed by an external source of neutrons produced by spallation reactions generated by a proton accelerator) are being conducted in collaboration with CNRS and European laboratories, among others, as part of the 5th, followed by the 6th European Programme for Research and Development initiatives. The results of the international programme TRADE (TRIGA Accelerator Driven Experiment) expected in 2008 will confirm the technological feasibility of the coupling of an accelerator and a subcritical reactor operating at power.

For transmutation studies in reactors able to produce electricity, in 1998, after the decision to stop the Super-Phénix breeder reactor was made, the experimental transmutation demonstration programme conducted in this reactor was redeployed to the Phénix reactor, which was bring it back up to power level on June 15, 2003.

Following results already obtained with Super-Phénix and Phénix (definition of families of new types of fuel suited for transmutation, development of americium targets in concentrated form), irradiations for transmutation studies will continue in Phénix in a much sustained way until 2008. The programme is intended to complete the qualification of the neutron data, test optimised irradiation matrices, study the behaviour under irradiation of specific fuel based on americium or other long-lived elements, and test advanced fuel concepts for the fourth-generation reactors. The first experiment MATINA 1 for testing of -material matrices get out of Phénix on February 2004.

Studies on fuel and transmutation targets also benefit from European and international cooperation. For example, in the HFR reactor at the European Joint Research Center in Petten, the full amount of americium contained in a target consisting of a magnesium oxide and aluminum was transmuted out.

Studies of scenarios continue in order to demonstrate the transmutation of actinides, in power reactor concepts of the fourth generation.

The fourth generation fast neutron systems have unique assets to recycle globally the Plutonium and the Minor Actinides, beyond the first recycles of Plutonium achievable in 2nd and 3rd generation PWRs.

Several scenarios of global management of the Plutonium and Minor Actinides produced in the PWRs have been analysed, depending on the date of deployment of the 4th generation systems.

These scenarios permit to cover reasonably the range of possibilities over the 21st century, while proposing a scenario consistent with the French generating fleet renewal strategy, and considering various alternatives corresponding to different options for the interim storage or the recycling of the Plutonium and Minor Actinides. The transition from today's fuel cycle in France to the deployment of Generation IV fast nuclear systems together with the continuing use of Generation III reactors lead to a mixed fleet of thermal and fast spectrum reactors to be optimised for managing globally the Actinides of both nuclear systems, and making best use of the available natural resources.

Conclusion

Advanced waste management strategies include the transmutation of selected nuclides, cost-effective decay-heat management, flexible interim storage, and customised waste forms for specific geologic repository environments. These strategies hold the promise to reduce the long-lived radiotoxicity of waste destined for geological repositories by at least an order of magnitude. This is accomplished by recovering most of the heavy long-lived radioactive elements. These reductions and the ability to optimally condition the residual wastes and manage their heat loads permit far more efficient use of limited repository capacity and enhances the overall safety of the final disposal of radioactive wastes.

Important results are now available, concerning the possibility of significantly reducing the quantity and the radiotoxicity of long-lived waste, concerning the feasibility of geological disposal and also relating to the modes of waste conditioning and long-term interim storage facilities. Complementary R&D work to be conducted in the coming years will enrich and consolidate these results and, in particular, will detail factors related to the industrial feasibility (costs, tentative schedule for implementation, etc.). On this basis, it can be assumed that sufficient technical and economic data will be available for decisions to be made in 2006 on the possible modes of long-lived waste management.

In this perspective, it is important to underline that:

- Waste which has been produced in the past, nowadays has to be disposed (Andra conducts the French research on this aspect).

- Plutonium control, recycling and management are the first priority in any strategy to minimise waste radiotoxicity.

- Processing and recycling are already significantly reducing the radiotoxicity of the ultimate radioactive waste. Furthermore, the further partitioning of minor actinides should reduce to a few hundred years the time necessary for the radiotoxicity level of vitrified waste to reach that of the natural uranium ore used to originally generate the waste.

- Key design criteria (closed cycle, recycling and integral consumption of actinides, fast spectrum) for the fourth generation of nuclear energy systems are set so that they can transmute their own long-lived waste and contribute to the transmutation of long-lived radionuclides (after their partitioning) produced in existing reactors.

- Despite this prospect of a very large reduction in the quantity and radiotoxicity of waste produced in nuclear power reactors, it will not be possible to get rid of ultimate waste which will have to be disposed.

- Last, the possibility of storing radioactive materials in a safe and robust fashion throughout the long-term is now confirmed. This provides a useful flexible tool for the implementation of back-end fuel cycle management strategies.

- These scientific achievements must be accompanied by a democratic and political debate under the care of government.

R&D ACTIVITIES FOR PARTITIONING AND TRANSMUTATION IN KOREA

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Abstract

According to the Korean long-term plan for nuclear technology development, KAERI is conducting a R&D project on the partitioning and transmutation (P&T) of long-lived radionuclides. The study for the partitioning is focused on the development of pyroprocessing based on an electrorefining of actinides because it is a kind of proliferation-resistive technology, where all the transuranic metals are separated together as a mixture. The major experimental items of this study include an electrorefining, electrowinning, cadmium evaporation and a molten salt waste treatment. Various behaviours of the electrodeposition of uranium, rare earths, alkali and alkaline earths in the LiCl-KCl electrorefining system have been examined through fundamental experimental work. As liquid cadmium was employed as the second cathode in this study, its removal by an evaporation, thus leaving transuranic as the final product, was examined. Treatment of the molten salt waste was also investigated by introducing zeolite for the immobilisation of the molten salts. As for the transmutation system, KAERI is studying the HYPER (HYbrid Power Extraction Reactor), a kind of subcritical reactor which will be connected with a proton accelerator. Up to now, a conceptual study has been carried out for the major elemental systems of the subcritical reactor such as the core, transuranic fuel, long-lived fission product target, and the Pb-Bi cooling system, etc. In order to enhance the transmutation efficiency of the transuranic elements as well as to strengthen the reactor safety, the reactor core was optimised by determining its most suitable subcriticality, the ratio of the height/diameter, and by introducing the concepts of an optimum core configuration with a transuranic enrichment as well as a scattered reloading of the fuel assemblies.

Partitioning of actinide by pyroprocessing

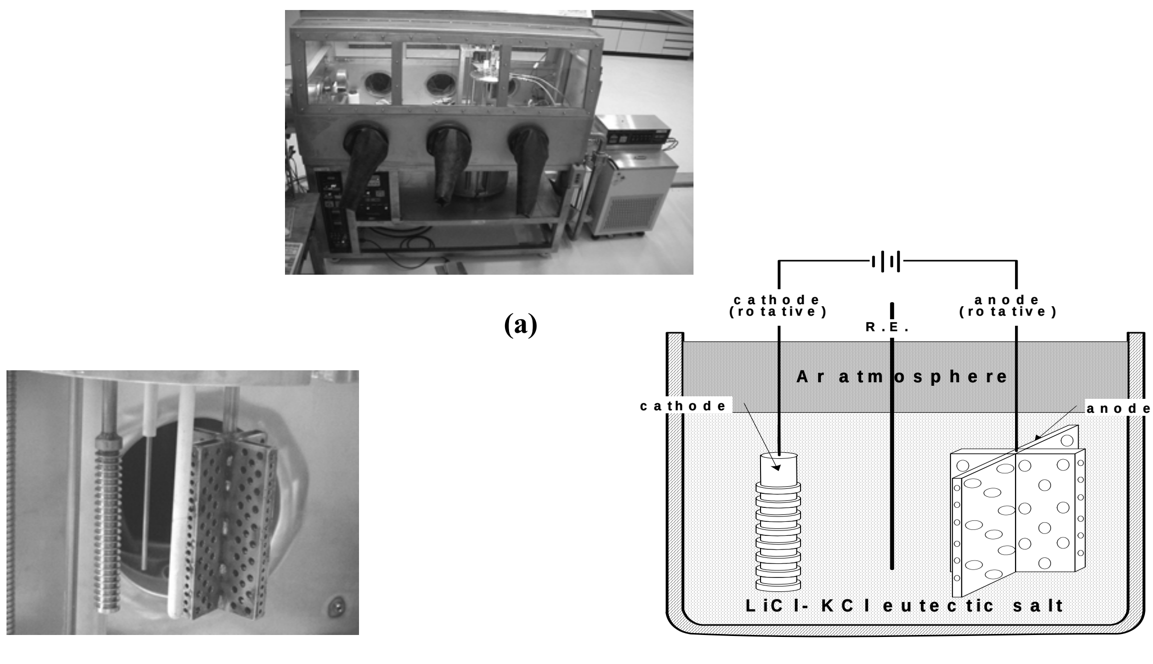
Removal of uranium by electrorefining

Since uranium is not desirable as a fuel source in the transmutation system, it is necessary to remove it from the spent fuels or high-level wastes in advance before recovering the transuranic elements to be transmuted. Electrorefining is a candidate technology to remove the uranium and then to recover the transuranic elements effectively. Since each metal chloride has a unique value of the Gibbs free energy of formation in the molten salt electrolysis, uranium can be selectively reduced and deposited onto the surface of the cathode by the adjustment of the electrical potential between the anode and cathode.

Figure 1 shows the experimental setup of the recent electrorefining system of KAERI. Electrolysis cell consisted of the anode basket containing the metallic fuel, screw type solid cathode and the reference electrode. The electrolysis cell and glove box are filled with an inert gas in order to protect the molten salt from being reacted undesirably with oxygen or moisture. The off-gas from the electrolysis cell is washed with an alkali solution and then water so that the chlorine and hydrochloride, that may be formed in the electrolysis, can be absorbed into the liquids. Some evaporated compounds from the molten salt can also be trapped in the liquids.

Figure 1. **The experimental setup for the molten salt electrolysis**

Cathode Anode

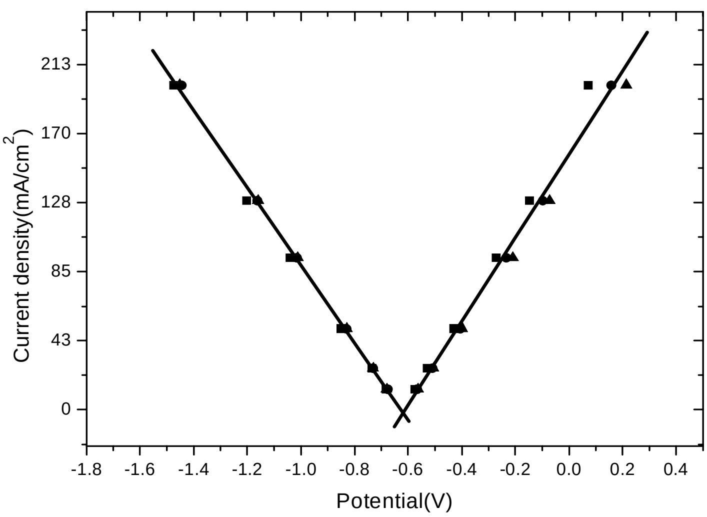


(b) **(c)**

(a) glove box system equipped with electrorefiner
(b) anode basket and screw type cathode
(c) electrode configuration

In this work, such operation variables as the rotation speeds of the anode and cathode, structure of the electrode, the initial concentration of uranium in the molten salt, deposited uranium morphology, current density, etc., affecting the electrodeposition of uranium were examined. Among the various experimental work, a typical polarisation curve for uranium was obtained as shown in Figure 2. It shows the relationship between the current density and the electrode potentials during the electrolysis. In general, alloy metals containing transition elements are used as the anode basket of the electrorefiner. However, it has been known that corrosion in the anode basket takes place when the potential is greater than 0.4 V for those materials. Accordingly, in order to prevent the anode basket from corrosion during the electrorefining, the current density should be kept at less than 220 mA/cm^2. A typical uranium deposition on the solid cathode at 100 mA/cm^2 shows a well developed dendrite structure.

Figure 2. **Polarisation curves for uranium at the anode and the cathode**



Recovery of the transuranic elements

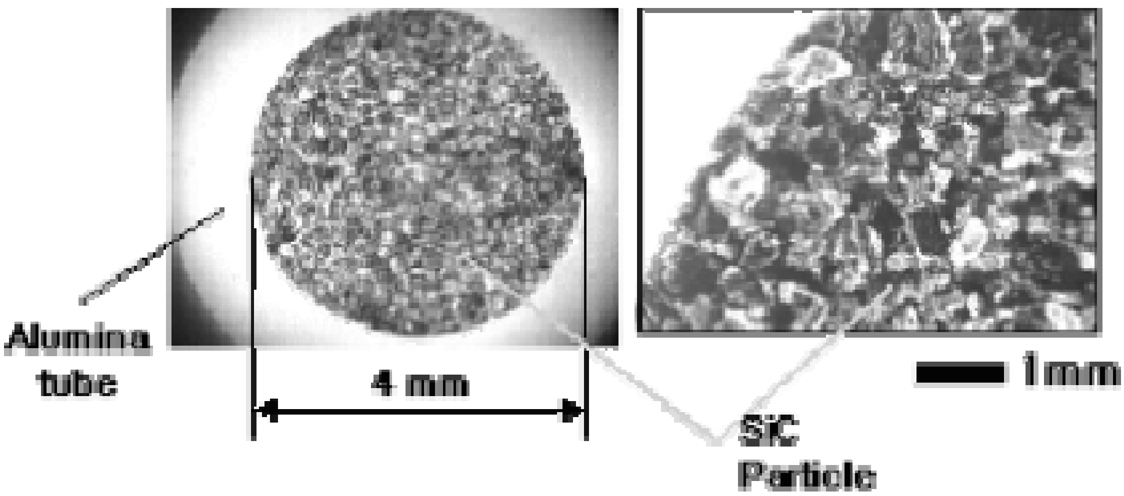
In this work, such rare earth elements as Nd, Ce, and La were used as a surrogate for TRU because they are very close to the TRU elements in the value of the Gibbs free energy of formation in the LiCl-KCl molten salt system. As a result, most amount of the rare earth elements contained in the molten salt (2.8 wt%) were deposited into the moderately agitated liquid cadmium which was used as another cathode. However, most of the deposits were formed at the interface between the salt and cadmium. It seems to be due to the fact that the rare earth elements have a low solubility in liquid cadmium and their density is smaller than that of cadmium but larger than that of the LiCl-KCl salt.

Further studies are also being carried out for the selective oxidation of the rare earth elements that exist in the cadmium phase and then the extraction of them into the molten salt phase in order to find its applicability for the removal of the rare earth elements from the TRU co-deposited into the cadmium.

Preparation of the reference electrode

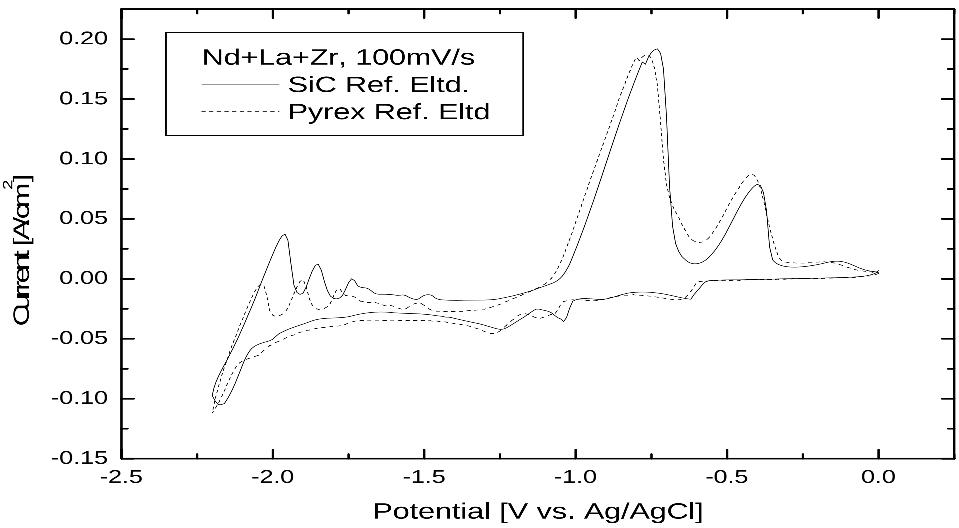
Since the existing reference electrode of Pyrex glass containing silver/silver chloride is quite fragile for handling during the experiments, we tried to develop a solider electrode for a more stable handling. In this work, we replaced the glass tube with an alumina tube and the glass membrane with a silicone carbide (SiC) membrane and then filled the tube with LiCl-KCl eutectic salt containing 1.0 wt% silver chloride where a silver wire was immersed and fixed as a tip of the electrode. Figure 3 shows the cross section of the SiC membrane which was prepared in our laboratory. The powders of SiC and a sintering binder were mixed together, placed at one end of the alumina tube and then sintered at 1 200 for 12 hours so that a porous membrane was formed and clogged the end of the tube. The pore size of this membrane was measured to be about 30 micrometers by means of a mercury porosimeter.

Figure 3. **The reference electrode made of alumina tube and SIC membrane**



The result of the cyclic voltammogram for the new electrode was compared with that obtained for the glass electrode in Figure 4, showing a consistant result in both cases.

Figure 4. **Cyclic voltamograms for the alumina/SiC and Pyrex type reference electrodes**



Design and analysis of the HYPER

Core

HYPER is designed to transmute TRU and some fission products such as 129I and 99Tc. HYPER is a 1 000 MW_{th} system and its k_{eff} is 0.98. Figure 5 shows a schematic configuration of the HYPER core with 186 ductless hexagonal fuel assemblies. As shown in Figure 6, the fuel blanket is divided into 3 TRU enrichment zones to flatten the radial power distribution. In HYPER, a beam of 1 GeV protons is delivered to the central region of the core to generate the spallation neutrons. To simplify the core design, the LBE coolant is used as a spallation target as well. In addition to the ultimate shutdown system (USS), six safety assemblies are placed in the HYPER core for an emergency case. The safety rods are also used conditionally to control the reactivity of the core. For a balanced transmutation of both TRUs and LLFPs (99Tc and 129I), 99Tc and 129I are incinerated in the moderated LLFP assemblies loaded in the reflector zone.

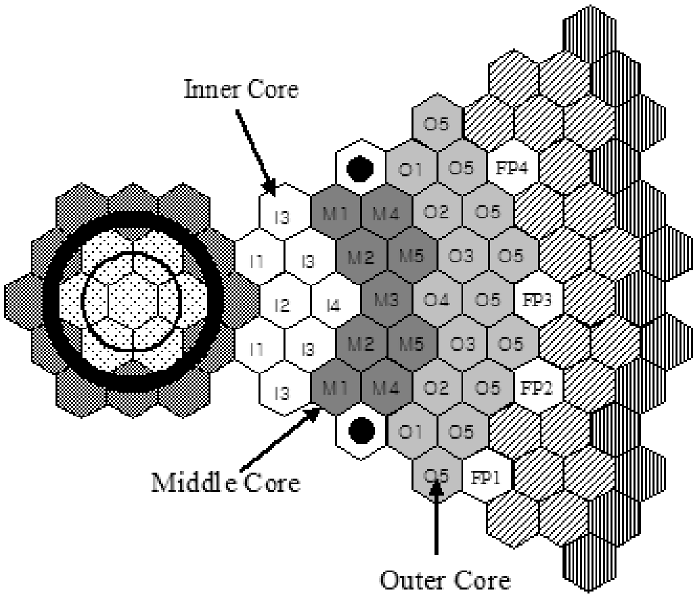
It is well-known that the LBE coolant speed is limited (usually < 2 m/sec) due to its erosive and corrosive behaviour. Therefore, the lattice structure of the fuel rods should be fairly sparse. In fast reactors, a pancake-type core has been typically preferred mainly to reduce the coolant pressure drop. Unfortunately, it has been found that the multiplication of the external source is quite inefficient in a pancake type ADS because of the relatively large source neutron leakage. It was shown that the maximum source multiplication can be achieved when the core height is about 2 m [1]. Taking into account the source multiplication and the coolant speed, the core height of HYPER was compromised at 150 cm, and the power density was determined such that the average coolant speed could be about 1.65 m/sec. To reduce the core size and improve the neutron economy, a ductless fuel assembly is adopted in the HYPER system. An advantage of the ductless fuel assembly is that the flow blockage of a subassembly is basically impossible and the production of activation products in the duct can be avoided.

Concerning a TRU-loaded ADS using a fixed cycle length, one of the challenging problems is a very large reactivity swing, leading to a large change of the accelerator power over a depletion period. Even in an ADS loaded with a MA (Minor Actinide) fuel, the burnup reactivity swing is found to be fairly noticeable, although it is relatively smaller than that in a TRU-loaded core. The large burnup reactivity swing results in several unfavorable safety features as well as deleterious impacts on the economics of the system. In the HYPER core, the B-10 was also used as a burnable absorber (BA) in a unique way to reduce the reactivity swing and control the core power distribution [2].

The required current is 10.6 mA at BOC and 16.4 mA at EOC. The inventory of TRU is 6 510 kg at BOC and 282 kg of TRU is transmuted per year. In the case of fission products, 129I and 99Tc are transmuted with the rates of 7 and 27 kg/yr respectively. The fuel cycle is 180 days. HYPER adopts a scattered fuel reloading system.

MC-CARD, REBUS-3 and DIF3D are used for the core analysis. The LAHET code system is used for the target neutronic calculations. KAERI also developed a kinetics code called DESINUR (Design Evaluation and Simulation of Nuclear Reactor).

Figure 5. **Schematic diagram of the HYPER core**

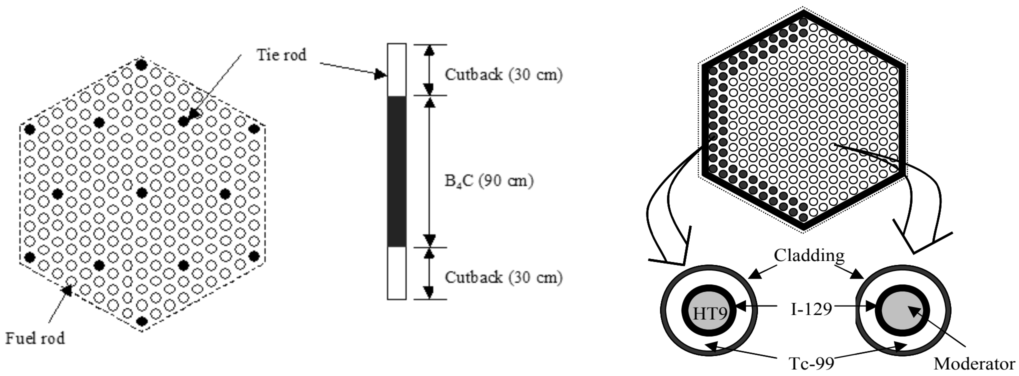


Fuel and Fission Product

In general, a non-uranium alloy fuel is utilised in a TRU transmuter to maximise the TRU consumption rate. Previously, a Zr-based dispersion fuel was used as the HYPER fuel since it was expected that a very high fuel burnup could be achieved. However, we have found that the dispersion fuel transforms to a metallic alloy during a high temperature operation. Therefore, in the current design, a metallic alloy of U-TRU-Zr is utilised as the HYPER fuel, in which pure lead is used as the bonding material. As a result, a large gas plenum is placed above the active core.

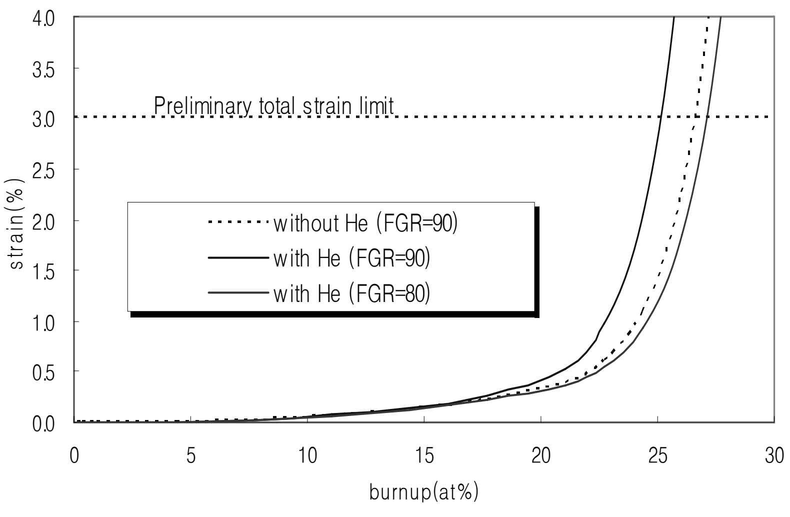
Each fuel assembly has 204 fuel rods and the fuel rods are aligned in a triangular pattern with 13 tie rods. A fairly open lattice with a pitch-to-diameter (P/D) ratio of 1.49 is adopted in HYPER. In Figure 6, a schematic configuration of the ductless fuel assembly is shown. The B-10 burnable absorber is loaded into the tie rods with top and bottom cutbacks in order to enhance the B-10 depletion rate and also to flatten the axial power distribution of the core. The BA concept with the cutbacks can effectively mitigate the peak fast neutron fluence of the assembly. The peak fast neutron fluence is a limiting design criterion in the LBE-cooled fast reactors.

Figure 6. **Fuel and fission product assemblies**



The MACSIS-H for an alloy fuel and the DIMAC for a disperison fuel are being developed as the steady-state performance analysis code, respectively. Main structures of each code consist of the temperature profile calculation routine, the swelling/FGR calculation routine, and the deformation calculation routine. The He production rates calculated by the other code are inserted into the swelling/FGR routine of each code. Figure 7 is one example of the MACSIS-H calculation. The strain was calculated as a function of the burnup of the HYPER fuel. The calculation was performed with and without a He generation by varying the fission gas rate. The result shows that the maximum strain is lower than the limit of the HYPER average discharge burnup of 17 at %.

Figure 7. Strain vs burnup for the HYPER fuel

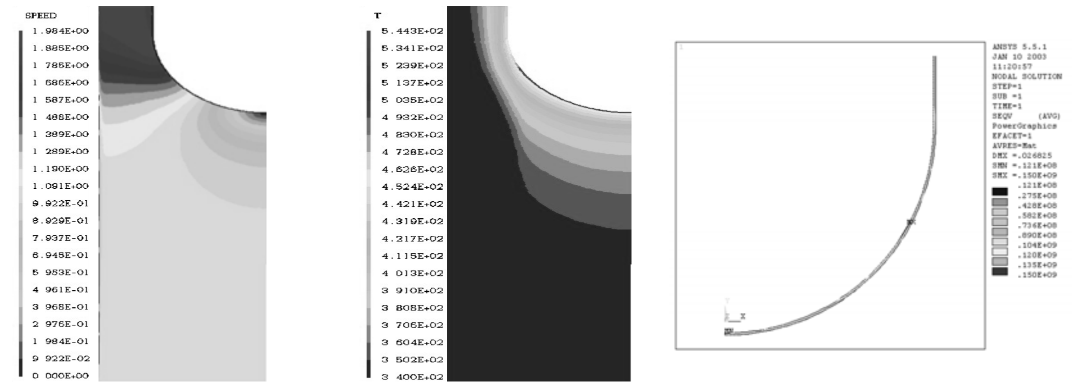


Coolant and Target

Pb-Bi is used as the coolant and spallation target material. The coolant is not separated from the target. MATRA and SLTHEN are used for the core thermal-hydraulic calculations. SLTHEN can be used for a multi-assembly analysis. Therefore, a 7 and 45 assembly analysis of HYPER was performed using SLTHEN. MATRA was developed to be used for the ductless assembly with a grid spacer, which is the case of the HYPER fuel assembly. The sub-channel analysis was performed using MATRA and the result shows that the average outlet temperature is 490^{o}C when the inlet temperature of the coolant is 340^{o}C. The maximum cladding temperature turned out to be 570^{o}C.

The cylindrical beam tube and the hemispherical beam window were adopted in the basic target design concept with a 1 GeV proton energy, and the thermal hydraulic and structural analyses were performed with the CFX and ANSYS codes. The target window material is 9Cr steel such as T91 and 9Cr-2WVTa. The beam window diameter and thickness were varied to find the optimal parameter set based on the design criteria: maximum Lead-bismuth eutectic (LBE) temperature < 500^{o}C, maximum beam window temperature < 600^{o}C, maximum LBE velocity < 2 m/s, and the maximum beam window stress < 160 MPa. The results show that a 40 cm wide proton beam with a uniform beam profile should be adopted for HYPER. It was found that a 2.5 mm thick beam window is needed to sustain the mechanical load. When the inlet velocity of Pb-Bi is 0.95 m/s, the maximum allowable current is 24.1 mA, which is greater than the required current of HYPER.

Figure 8. **CFX and ANSYS results of the HYPER target calculation**



Velocity (Max=1.96 m/s) | Temperature (Max=544 ^{o}C) | Stress (Max=150 MPa)

Experimental

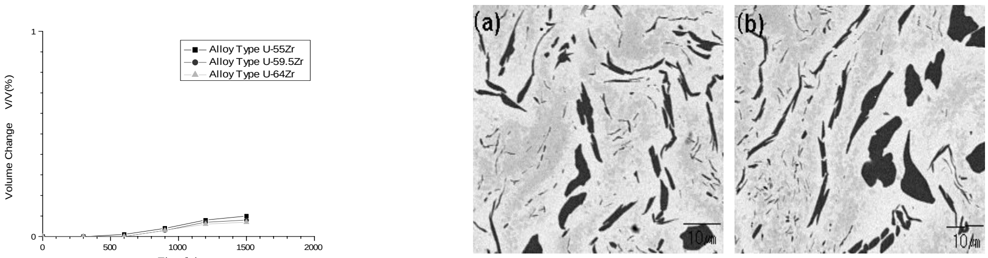
Fuel

Two types of metal fuels were considered for the HYPER fuel. One is a metal alloy type and the other is a dispersion type. Both types of U surrogate fuel samples were fabricated and the basic characteristics were investigated. It was found that the dispersion fuel was not made with a good microstructure and the original structure of the dispersion fuel was not kept after an annealing. Therefore, we chose the metal alloy fuel for the HYPER fuel type.

The reference blanket fuel pin of HYPER consists of the fuel slug of the TRU-xZr (x=50-60wt%) alloy and it is immersed in lead for a thermal bonding with the cladding. The blanket fuel cladding material is ferritic-martensitic steel HT-9. As a basic study on the HYPER fuel, we fabricated surrogate U-50, 55, 60wt%Zr alloy fuel instead of the actual TRU-Zr fuel. The U-Zr metallic fuel was fabricated by mixing, pressing, sintering and extrusion. The sintering temperature and time were 1500°C and 2 hrs respectively.

After fabricating the surrogate fuel, the thermal properties such as the thermal conductivity and the thermal expansion coefficient were measured. We also performed a thermal stability test of the surrogate fuel to investigate the volume change and the microstructure change. The fuel samples were put into the furnace for 1 500 hrs with the temperature of 630°C and 700°C. Figure 9 shows the volume change as a function of the time and the microstructure change after an annealing. Reaction characteristics were also investigated among the Pb bonding, cladding material (HT-9) and U surrogate fuel.

Figure 9. **Volume and microstructure change of the U-Zr sample after an annealing**



Volume change

Microscopic structure : (a) Before (b) After

Pb-Bi

KAERI joined the MEGAPIE project in 2001 for the experimental study of Pb-Bi. MEGAPIE is the one megawatt proton beam irradiation test of the Pb-Bi target. PSI, CEA, CNRS, FZK, ENEA, SCK-CEN, KAERI, JAERI and LANL are members of the MEGAPIE project. Now, the Pb-Bi target is in the stage of fabrication.

The most significant problem in handling Pb-Bi is corrosion. Therefore we performed static corrosion tests using FZK's facility COSTA to investigate the dissolution effect. KAERI also installed a static corrosion facility in 2003 and started static corrosion tests. Figure 10 shows the schematic diagram of the static corrosion facility. It is mainly composed of tube furnaces, a gas system and a glove box. The furnace has three independent zone heaters to reduce the temperature difference.

The test materials were 316LN and some ferrite/martensitic steels such as HT-9 and T91. The test was performed under both reduced and oxygen-controlled atmospheres. Part of the test results are shown in Figure 10.

Figure 10. **Schematic diagram of the KAERI static corrosion facility and test results**

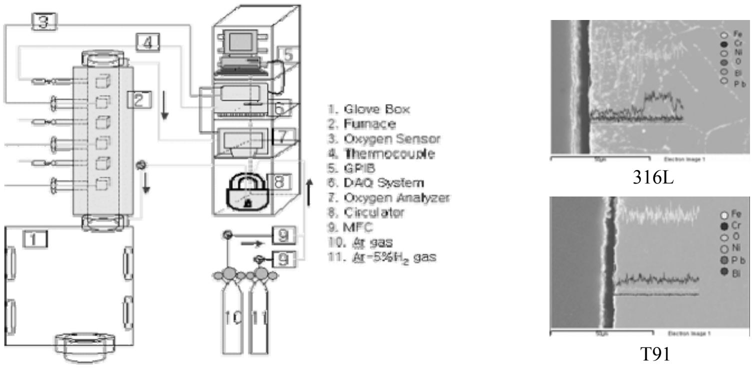
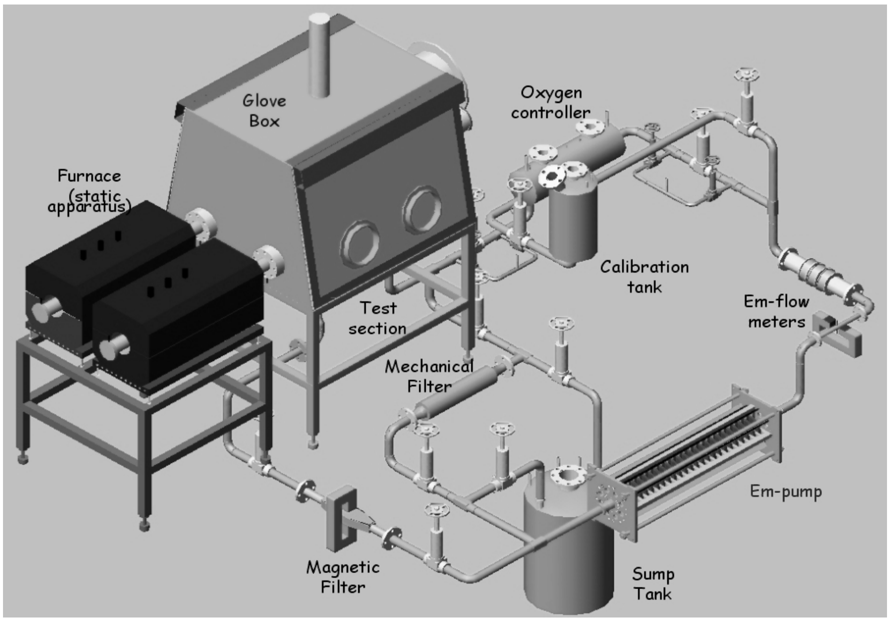


Figure 11 shows the schematic diagram of the dynamic corrosion loop to be installed at KAERI. The LBE loop is an isothermal loop. The flow velocity in the test section was designed to be around 2m/s in the range of 400 ~ 550°C and the charging volume of the LBE is around 0.03 m^3 in the circulation loop.

Figure 11. **Three dimensional schematic diagram of the corrosion loop**



The LBE loop is mainly composed of a main test loop, bypass loop for filtering the LBE and a mixture gas supplying system. The liquid metal in the main test loop circulates in the following order: EM pump – EM flow meter – oxygen controller – test section – magnetic filter – EM pump. From the analysis of the pressure drop, the specification of the piping system was determined as a 1.5 inch pipe to reduce the pressure drop by a high mean fluid velocity. The pressure drop of the main test loop was estimated at around 3 bar with the flow rate of 60 lpm.

The oxygen concentration in the range of 10^{-7}wt% ~ 10^{-5}wt% is controlled by the chemical equilibrium between the mixture gas of hydrogen-argon and the water vapor. At present, the oxygen concentration in the LBE and the mixture gas is measured with an oxygen sensor made of Yttria Stabilised Zirconia as a solid electrolyte cell and Pt/air as a reference system.

Summary

KAERI has been working on ADS since 1997. The KAERI ADS system is called HYPER (HYbrid Power Extraction Reactor). HYPER research started as a 10 year nuclear research programme. The ADS research of KAERI consists of three stages. The conceptual design of the core was almost completed in the second stage (2000-2003). The core design will be upgraded and modified in the third stage (2004-2006). For example, the dual annular injection tube will be introduced to reduce the flow rate of the Pb-Bi in the target channel. The structure analysis of the HYPER fuel assembly will be also performed. But the main work related to the design and analysis is a transient case study. Some core transient cases such as LOHS-WS and LOF-WS will be studied. The transient cases related to the target, beam window and fuel assembly structure will also be studied. KAERI fabricated an U surrogate metal fuel and performed tests using the U fuel sample to investigate the basic characteristics in the second stage of research. The HYPER fission product target includes both 99Tc and 129I in the same rod. We will fabricate the fission product rod and test it using KAERI's research reactor HANARO, which is a 30 MW reactor. Fabrication study of the FP target will be performed in 2004 and the irradiation will start in 2005. KAERI will complete the construction of a corrosion loop in 2004. The oxygen control method is considered to test the protection of steel structure materials against Pb-Bi corrosion. Therefore, an oxygen sensor will be developed. KAERI will launch the I-NERI programme in June, 2004. The lead-alloy corrosion will be investigated through the I-NERI programme. The period of I-NERI is three years and LANL is the U. S. partner.

Acknowledgements

The Korean Ministry of Science and Technology (MOST) has supported this work.

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OVERVIEW OF RUSSIAN P&T PROGRAMME

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Abstract

The status of Russian programmes of RW handling is outlined. R&D programmes of RW incineration are presented, particularly:

- RW transmutation in reactors;
- ADS test facility SAD (Dubna);
- Project of ADS facility at Moscow Meson Factory (Troitsk);
- Design study of the cascade subcritical molten salt reactor-burner;
- Programme of comparative study of the different types of ADS-burners;
- Programme of systematic study of the minor actinide nuclear characteristics: measurements and evaluation.

Introduction

The strategy of the nuclear power in Russian Federation is the closed nuclear fuel cycle (CNFC). The essential feature of this strategy is the principle of the radiation equivalence of the row uranium from mines and buried radioactive waste (RW) [1-4].

At present, the total amount of the nuclear spent fuel in storage of Russia is ~16 thousand tons. It grows annually for 850 tons and only part of them (~ 20%) is directed to RT-1 plant for the regeneration. The conception of "delayed decision" is accepted for the other spent fuel as well as the additional spent fuel from the new nuclear power plants under construction. It is accepted that 50 years is enough to develop the industry of RW handling and transmutation and to organise CNFC.

RW handling

The streamline of the RW handling activity today is preparation of the spent fuel storage which will be able to keep RW safely during next 50 years.

The PUREX-technology is used now for the spent fuel reprocessing but several new ones were suggested and studied:

- gas-fluoride technology, which reduces the amount of solid high level FP up to 1 m^3 for every 1 ton of the reprocessed spent fuel and allows to extract U from the spent fuel without Pu separation [5];

- pirochemical reprocessing which allows to simplify essentially the spent fuel reprocessing and to reduce the amount of low level FP [6].

Some others R&D in this line are presented in the Proceedings of the recent conference on the radioactive waste management (St. Petersburg, 2004) [7].

RW transmutation

Two main strategies of RW transmutation are studied today in Russia: transmutation in critical reactors and in ADS-burners.

Reactor transmutation

In the BREST conception (fast reactor with the lead coolant) developed at RDIPE [8] the RW – transmutation takes place in the active core of reactor. The fuel composition of this reactor consist of the spent fuel of thermal reactors cleaned from fission products (FP) up to level 1÷5% only and enriched by Pu and minor actinides (MA) up to level ~ 10% and $\lesssim$ 5% respectively. Such a fuel composition does not need in the subsequent Pu regeneration and allows to burn-up annually the MA-production of several thermal reactors of the same power as BREST reactor. Isotopes ^{99}Tc and ^{129}I are incinerated in the thermal blanket (~ 250 kg per year in reactor BREST-1200). The radioactive equivalence in such a schema can be achieved in 80 years if buried FP will be purified from MA to the level ~ 10^{-3}. ^{90}Sr and ^{137}Cs are buried separately: their radioactive equivalence is achieved in ~ 200 years [9-11].

Experimental study of MA burning in the fast neutron spectrum of reactor BOR-60 was carried out in the framework of DOVITA-programme during the last ten years at RIAR (Dimitrovgrad) [12-14]. On the basis of these researches the actinide burner fast reactor (ABFR) was suggested. The experimental study of MA transmutation has been performed also at BFS-stand (IPPE, Obninsk) [15].

In the transient multicomponent nuclear power the problem of RW-transmutation should be solved using special critical reactors or ADS-burners.

The transmutation capability of the Na-cooled fast reactor BN-800 was studied at IPPE (Obninsk) [16-17]. It was shown that one reactor BN-800 in principle can transmute MA from 5-6 thermal reactors of the same power.

The conception of RW-burning in the heavy water reactor is developed at ITEP (Moscow) [18-19]. The authors insist that the transmutation potential of thermal reactors is not yet studied properly.

ADS transmutation

Large scale nuclear power has a future on the fast reactor basis only. RW-transmutation problem in this case will be solved naturally as a byproduct of the nuclear power generation.

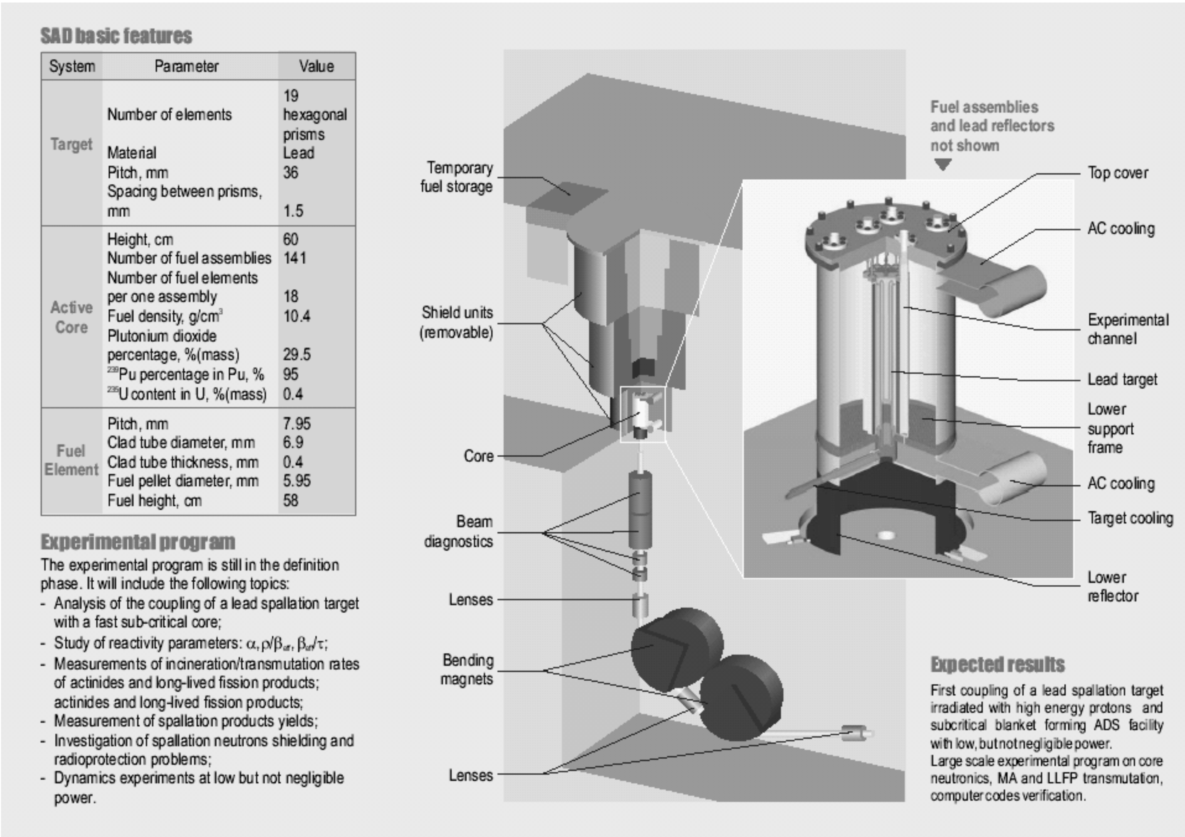
But in the transient phase of nuclear power development or in the case if society will decide to stop the nuclear industry, ADS is the most reliable reactor-burner due to the safety reasons: the small fraction of MA delayed neutrons and bad Doppler-effect in the absence of ^{238}U prevent the creation of the safety critical reactor-burner.

There are several works in this line in Russia outlined below:

Subcritical assembly at Dubna (SAD)

The design study of ADS-test facility SAD is completed and SAD construction has started at JINR (Dubna) this year [20]. SAD layout is presented on Figure 1. The experimental programme consist of the measurements of subcriticality, MA fission rates, spallation product yields and studies of reactivity feedbacks (α-value, ρ/β_{eff}, β_{eff}/τ, etc.), as well as benchmark experiments for testing numerical codes.

Figure 1. **Layout and basic parameters of SAD**

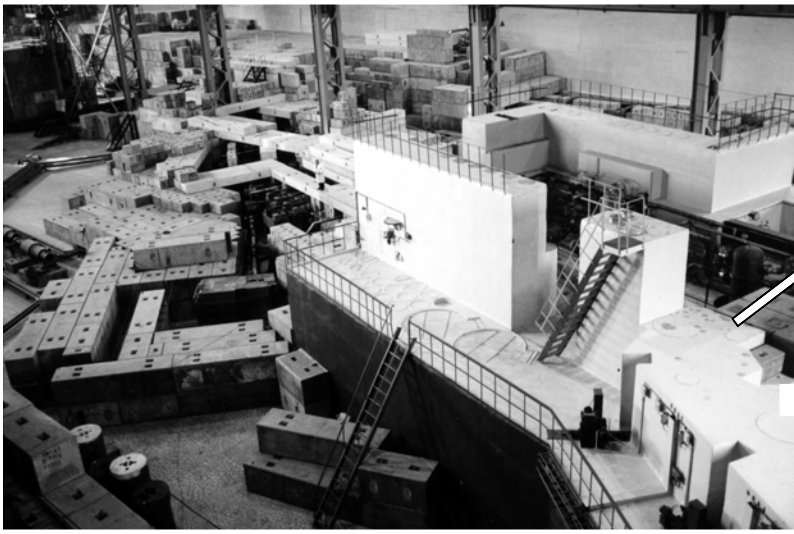


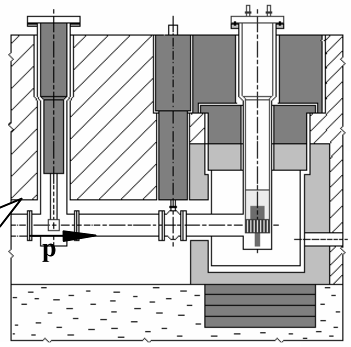
SAD will be the first real ADS which contain all the essential components: proton accelerator with beam power ~ 1 kW (energy 680 MeV, current ~3 μA), spallation lead target and fast spectrum subcritical core with power ~ 20 kW (U with ~ 30% Pu contamination).

ADS facility based on MMF-linac

Moscow Meson Factory (MMF, Troitsk) has all the infrastructure for using it as ADS test facility with blanket thermal power ~ 5 MW. Upgrade project is prepared now and it is waiting for financing and interest from ADS-community (see Figure 2 and Table 1) [21].

Figure 2. **Experimental area of MMF and schema of ADS-target**





ADS-target

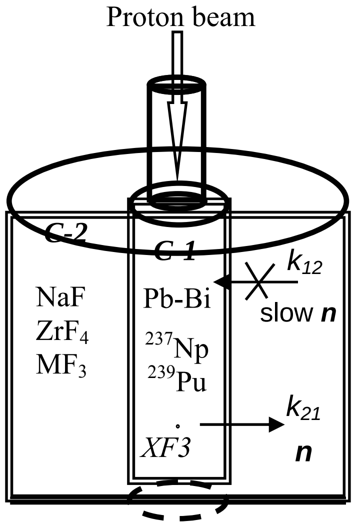
Table 1. **Basic characteristics of the planned ADS facility at INR**

Accelerator:	Proton energy	500 ÷ 600 MeV
	Average current	0.15÷0.3 mA
Target:	Material	Tungsten, other materials
	Thermal power	75÷150 kW
	Coolant	Water
Core:	Thermal power	3÷5 MW
	Coolant	Water and PbBi
	Fuel	Enriched Uranium, MA
	Neutron spectrum	Fast resonance

Cascade subcritical molten salt reactor (CSMSR

In the framework of ISTC project #1486, the design study of CSMSR-burner has been performed [22]. The essence of this reactor is cascade scheme of neutron flux amplification (Figures 3-4) which allows to reduce several times the power of accelerator-driver [23]. The parameters of CSMSR-burner have been estimated [24-26].

Figure 3. **Cascade schema of CSMSR: the back thermal neutron flux from Core-2 to Core-1 is suppressed (k_{12}≈0)**

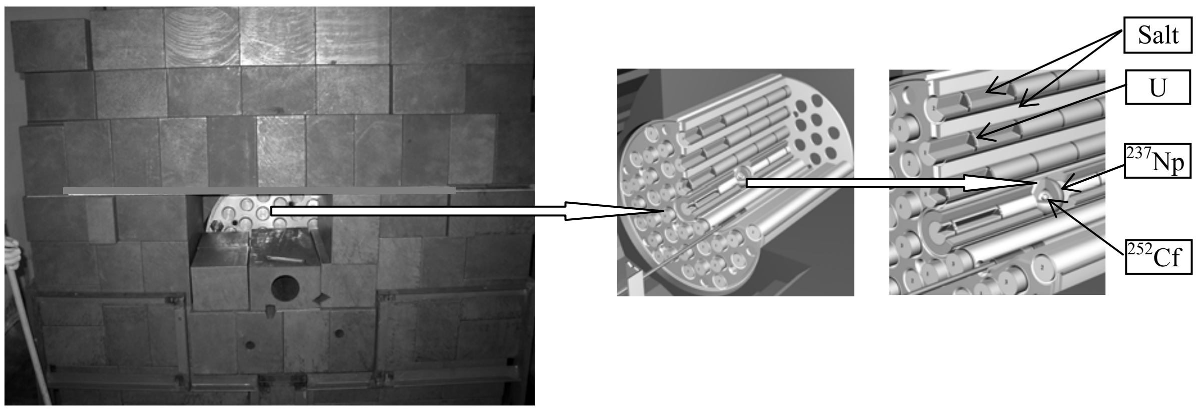


Subcriticality $\Delta k=1-k_{eff}$ of cascade two core assembly is

$$\Delta k = \frac{\Delta k_1 + \Delta k_2}{2} - \sqrt{\left(\frac{\Delta k_1 - \Delta k_2}{2}\right)^2 + k_{12}k_{21}} \ ,$$

where Δk_1 and Δk_2 are subcriticalies of cores C-1 and C-2.

Figure 4. **Experimental setup for study of the neutron flux cascade amplification**



Comparative study of different types of ADS-burners

There are three main approaches to RW transmutation via ADS-burner:

- heavy metal ADS (fast neutron spectrum);
- molten salt ADS (intermediate spectrum);
- heavy water ADS (thermal spectrum).

Everyone of this ADS-burners has its own advantages and shortages but up to date there is no their systematic comparative study on the common basis of uniform criteria.

Last year Minatom RF has approved the programme of the comparative study of all these types of ADS-burners (burning efficiency, MA loading, MA/Pu ratio, reprocessing features, etc.).

Systematic study of MA nuclear characteristics

R&D of MA transmutation by any method need as a first step the reliable data of MA nuclear characteristics. At the moment the precision of these data is not sufficient [27-31]. The programme of systematic measurements and evaluation of nuclear characteristics of all MA isotopes in the energy range 0.05 eV÷30 MeV has been suggested recently [32]. It is planned to measure differential and integral characteristics of 22 isotopes of U, Pu, Np, Am, Cm, Bk and Cf (see Table 2) using the purified set of MA isotopes, qualified teams and unique installations: high flux reactors SM-3 and BOR-60 (RIAR, Dimitrovgrad), BFS stand (IPPE, Obninsk), linac LU-50 (VNIIEF, Sarov), tandem generator (IPPE, Obninsk), lead slow-down 100 t spectrometer (INR, Troitsk).

Cross-check of the integral and differential measurements as well as the data evaluation and preparation of files compatible with reactor codes will be done by the world known Russian team.

Table 2. **List of MA isotopes planned to be measured**

	Isotope	**Measuring characteristics**	**Energy range**	**Experimental installation**
1	^{237}Np	$\sigma_c(E)$, $\sigma_f(E)$, RIC	0.05÷12 MeV	*Linac*
		$\langle\sigma_f\rangle$*), $\langle\sigma_c\rangle$, $\langle\sigma_{in}\rangle$, $\Delta\langle\sigma_c\rangle/\Delta T$, *CRC* **)	0.05÷5 MeV	*BFS*
2	^{238}U	$\Delta\langle\sigma_c\rangle/\Delta T$, $\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.05÷100 keV	*BFS, SM-3*
3	^{238}Pu	$\sigma_c(E)$, $\sigma_f(E)$, RIA,	04÷12 MeV	*Linac,*
		$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷5 MeV	*BFS*
4	^{239}Pu	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$, CRC	1÷5 MeV	*BFS*
5	^{240}Pu	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$, CRC	Fiss.thresh.÷5 MeV	*BFS, SM-3*
6	^{241}Pu	$\alpha(E)$, RIF, RIC, $\sigma_f(E)$	0.5÷2000 eV	*Linac*
		$\langle\sigma_f\rangle$	1÷5 MeV	*BFS*
7	^{242}Pu	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷100 keV	*SM-3*
8	^{244}Pu			
9	^{241}Am	RP, RIF, RIA, $\sigma_f(E)$	0.05÷12 MeV	*Linac*
		$\sigma_f(E)$	5÷30 MeV	*Tandem generator*
		$\langle\sigma_f\rangle$, $\Delta\langle\sigma_c\rangle/\Delta T$, $\langle\sigma_c\rangle$, CRC	Fiss.thresh.÷5 MeV	*BFS*
10	^{242m}Am	RP, RIF, RIA, $\sigma_f(E)$	0.5÷12 MeV	*Linac*
		$\sigma_f(E)$	1 eV÷30 MeV	*Tandem generator*
11	^{243}Am	$\sigma_\gamma(E)$, RIF, RIA, $\sigma_f(E)$	0.05÷5 MeV	*Linac*
		$\sigma_f(E)$	1 eV÷30 MeV	*Tandem generator*
		$\langle\sigma_f\rangle$	Fiss.thresh.÷5 MeV	*BFS*
12	^{242}Cm	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷100 keV	*SM-3*
13	^{243}Cm	$\sigma_f(E)$	5÷30 MeV	*Tandem generator*
		$\sigma_\gamma(E)$, RP, RIF, RIA	0.5÷100 eV	*Linac*
14	^{244}Cm	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	Fiss.thresh.÷5 MeV	*SM-3, BFS*
		$\sigma_f(E)$	5÷30 MeV	*Tandem generator*
		$\sigma_f(E)$, $\sigma_\gamma(E)$, $\alpha(E)$ RIF, RIC	0.5÷200 eV	*Linac*

	Isotope	Measuring characteristics	Energy range	Experimental installation
15	^{245}Cm	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷5 MeV	*SM-3, BFS*
		$\sigma_f(E)$	5÷30 MeV	*Tandem generator*
		RP, $\sigma_a(E)$, $\alpha(E)$, $\sigma_f(E)$ RIF, RIA	0.5÷2 keV	*Linac*
16	^{246}Cm	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷100 keV	*SM-3*
		$\sigma_f(E)$	5÷30 MeV	*Tandem generator*
		RP, $\sigma_c(E)$, RIC	0.5÷50 eV	*Linac*
17	^{247}Cm	$\sigma_f(E)$	5÷30 MeV	*Tandem generator*
		$\sigma_a(E)$, RIF, RIA, $\sigma_f(E)$	0.5÷2 keV	*Linac*
18	^{248}Cm	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷100 keV	*SM-3*
		$\sigma_f(E)$	5÷30 MeV	*Tandem generator*
19	^{249}Bk	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷100 keV	*SM-3*
20	^{250}Cf			
21	^{251}Cf	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷100 keV	*SM-3*
		RP, RIF, RIA	0.5÷5 keV	*Linac*
22	^{252}Cf	$\langle\sigma_f\rangle$, $\langle\sigma_c\rangle$	0.1÷100 keV	*SM-3*

***)** - $\langle\sigma\rangle$ is the spectrum averaged cross-section.

****)** - *CRC* is the central reactivity coefficient.

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THE PROGRESS OF P&T ACTIVITIES IN CHINA

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Abstract

It could be foreseen that the total nuclear power capacity will reach 8.7 GWe before the year 2005, 14-15 GWe before 2010 and nearly 36 GWe before 2020 respectively in the mainland of China. The 65 MWt China Experimental Fast Reactor (CEFR) is under construction. The first criticality is expected at the end of 2007. As one project of “the major state basic research programme (973)” in energy domain, which is sponsored by the China Ministry of Science and Technology (MOST), a five-year programme of basic research for ADS physics and related technology has been launched since 2000. The main progress on CEFR, ADS and other related topics will be reported.

Introduction

China, as a developing country with a great number of population and relatively less energy resources, reasonably emphasises the nuclear energy utilisation development. Due to the economy and experience reasons, the nuclear power and technology development with a moderate style are kept in China up to now.

Presently, in the mainland of China, there are nine reactor units in operation at five NPP sites with the total capacity of 6.7 GWe; two units at one NPP site are under construction, and two NPP sites are planned within the frame of the Tenth Five-year Plan (2001-2005) as shown in Table 1. Also, another one or two NPP sites are still under discussion. It can be foreseen that the total nuclear power capacity will reach 8.7 GWe before the year 2005, 14-15 GWe before 2010, and nearly 36 GWe before 2020, respectively.

Table 1. **Mainland Nuclear Power Plants**

NPP	Type	Power (MWe)	Commercial operation
Qinshan-1	PWR	300	1993
Daya Bay	PWR	2×900	January and June 1994
Qinshan-2	PWR	2×600	April 2002 and May 2004
Qinshan-3	PHWR	2×728	December 2002 and July 2003
Lingao-1	PWR	2×984	May 2002 and March 2003
Lian yungang	PWR	2×1000	2004 and 2005
Sanmen	PWR	2×1000	2010 (to be expected)
Lingao-2	PWR	2×1000	2010 (to be expected)
Yangjiang	PWR	2×1000	2010 (to be expected)

To develop nuclear power in large scale, two problems must be solved. First, as we understand the technically and economically exploitable natural uranium resources are limited domestically or overseas, so the uranium utilisation rate has to be raised greatly. Second, long-lived radioactive nuclear wastes have to be in disposal to reduce its impact to environment and public fear to nuclear power.

Right now only small amount of spent fuels from NPPs has been accumulated in China. But the situation will be very serious in the future according to above prediction of nuclear energy development in China. The annual generation of waste is estimated to 2 275 7 500 and 10 000 m^3 respectively for the year 2004, 2010 and 2020.

Considering MA and LLFP transmutation with more efficiency and non-criticality risk for new nuclear application, the fast breeder reactor technology and the accelerator-driven sub-critical system (ADS) have been started to develop as a national research projects in China.

Fast reactor technology development in China [1-3]

For the continuously development of nuclear energy and the effective utilisation of the uranium resource, the development of fast breed reactor in China was started early in 1965. The basic researches for FBR in China could be divided into two Phases. The activities from the middle-end of 60s to the year 1987 focus on fast neutron physics, thermohydraulics, materials, sodium technology and some sodium components in small size. During this period about 12 sodium loops and test facilities have been built up including a fast neutron zero power facility DF-V containing 50 kg ^{235}U.

From the year 1987 to 1993, all research activities were arranged with a target of 65 MWt experimental fast reactor. The emphasis was put on fast reactor design study, sodium technology, and fast reactor safety, fuels and materials as well as sodium components. During this period about 20 sodium loops and testing facilities have been established and tested. After above period's researches the R&D of fast reactor technology are carried out as the design demonstration tests which more than 30 items.

The 65 MWt China Experimental Fast Reactor (CEFR) supported by the National High Technology Programme (863) is under construction. The conceptual design of the CEFR was started at 1990 and completed at 1993 including the confirmation and optimisation to some important design characteristics. Its preliminary design was completed in August 1997. The detail design was started since the early 1998. In May 2000 approval of the construction was issued and the first barrel of concrete was poured. The construction of reactor building (57 meters above the ground) with about 40 000 m^2 floor surface was completed in August 2002.

CEFR is a sodium cooled 65 MWt experimental fast reactor with $(Pu, U)O_2$ as fuel, but UO_2 as first loading, Cr-Ni austenitic stainless steel as fuel cladding and reactor block structure material, bottom supported pool type, two main pumps and two loops for primary and secondary circuit respectively. The water-steam tertiary circuit is also two loops but the superheat steam is incorporated into one pipe which is connected with a turbine. Table 2 shows the main design parameters of CEFR.

The final safety analysis report (FSAR) is being prepared. Much progress was made with regard to the topics required by the nuclear safety authority. Almost all the safety-related design demonstration tests have been carried out. The detailed design work is nearly completed (about 95%), except for the instrumentation and control (I&C) system. Ninety percent of the concrete constructions, including the main building, have been completed. About 300 components have been installed in the building. The steel liner and the ventilation pipes are being installed. Components and instruments made in domestic companies are fabricated smoothly and delivered to the site continuously, except for the main vessel, whose delivery is postponed by two years due to the delays in material procurements and the shortage of production capacity at the manufacturer. Some components and instruments imported from foreign companies, for example the control rod drive mechanisms (CRDM), the in-vessel fuel handling machines, the sodium valves, and others, have been delivered to the site.

Table 2. **CEFR Main Design Parameters**

Parameter	Unit	Preliminary design
Thermal Power	MW	65
Electric Power, net	MW	20
Reactor Core		
Height	cm	45.0
Diameter Equivalent	cm	60.0
Fuel		$(Pu, U)O_2$
Linear Power max.	W/cm	430
Neutron Flux	$N/cm^2 \cdot s$	3.7×10^{15}
Bum-up, target max.	MWd/t	100000
Bum-up, first load max.	MWd/t	60000
Inlet Temp. of the Core	^oC	360
Outlet Temp. of the Core	^oC	530
Diameter of Main Vessel (outside)	M	8.010
Primary Circuit		
Number of Loops		2
Quantity of Sodium	T	260
Flow Rate, total	t/h	1328.4
Number of IHX per loop		2
Secondary Circuit		
Number of loops		2
Quantity of Sodium	T	48.2
Flow Rate	t/h	986.4
Tertiary Circuit		
Steam Temperature	^oC	480
Steam Pressure	MPa	14
Flow Rate	t/h	96.2
Plant Life	A	30

The general programme, the quality assurance guideline, and the schedule for the pre-operation testing have been defined. The testing procedure, testing guideline and testing safety criteria for each system are under preparation. For the physics start-up, the test list and test methods have been determined, and the related instruments and equipments have been ordered. In particular, the fast neutron zero-power reactor DF-VI has been moved to Beijing and will be used for operator training, demonstration of the experimental neutron physics methods, as well as for neutron detector testing. Presently, 28 operators, including four senior operators, are being trained. Due to the delay in the delivery of the reactor main vessel, the project schedule has been updated. The first criticality, originally planned for the end of 2005, is now postponed to the end of 2007.

The 600 MWe Chinese Prototype Fast Reactor (CPFR) is being considered, and the submittal of the relative proposal to the government is being been considered.

Progress in ADS system research

The conceptual study of Accelerator Driven System (ADS) [4] had lasted for about five years and ended in 1999 in China. As one project of "the major state basic research programme (973)" in energy domain, which is sponsored by the China Ministry of Science and Technology (MOST), a five years programme of basic research for ADS physics and related technology has been launched since 2000. The research activities are focused on HPPA physics and technology, reactor physics of external source driven sub-critical assembly, nuclear data base and material study [5]. For HPPA, a high current injector consisting of an ECR ion source, LEBT and a RFQ accelerating structure of 3 MeV will be built. A 1 GeV/20 mA linac is in the conceptual study [6]. In reactor physics study, a series of neutron multiplication experimental study has been carried out and is being carrying on. Instead of the verification facility consisted of a 150 MeV/3 mA linac and a modified swimming pool light water reactor of 3.5 MW described in [4], a rather modest but more realistic facility is in consideration. CIAE (China Institute of Atomic Energy), IHEP (Institute of High Energy Physics), PKU-IHIP (Institute of Heavy Ion Physics in Peking University) and other institutions are jointly carrying on above mentioned research.

DongFeng 3 experiment

DongFeng 3 is an existing light water moderated zero power critical assembly with flexible core structure. We re-arranged the core-structure loaded with 20% enriched U_3O_8 fuel pin with 400 mm active length packed in φ6 aluminum tube of 1 mm thickness. On this arrangement, the nuclei number ratio N_H/N_5 is about 270. A strong ^{252}Cf neutron source of 2×10^9n/sec is used in the experiment. In central area of the core, there is the buffer. Water, lead and stainless steel are used as buffer material respectively. Experimental measurements of k_{eff} in sub-critical mode were carried out with different buffers by using source jerk and extrapolation-period method. The measurements of spatial distribution of fission rate and neutron flux have been done by means of a solid state nuclear track detector (SSNTD). The measurements were carried out both in critical and sub-critical modes with ^{252}Cf neutron source with different buffer materials. In sub-critical mode, the source position was at the center of the core and at the button of the structure respectively.

Venus I experiment

A composed structure of zero-power sub-critical assembly combined with a pulsed neutron source, Venus I programme is being carrying on followed the DongFeng 3 experiment. The pulsed-neutron will be provided by a Cockroft-Walton machine, routinely operated since 2001. Fourteen MeV and 2.5 MeV neutrons will be derived by d-T and d-D reaction. The neutron yield in DC mode can reach 10^{12}n/s, while in micropulse mode 10^9n/s ~ 10^{10}n/s for d-T reaction.

There is a source and buffer in the centre, a driven zone consisted of natural Uranium pin is very densed lattice with aluminum in between, an active zone with 20% and mainly 3% enriched ^{235}U fuel pin is polyethylene lattice and the polyethylene reflector. Different neutron spectra in different zone are expected. The buffer will shift the sharp 14 MeV and 2.5 MeV neutrons to the fast neutron spectra to mock-up the evaporation bump in the spallation neutron spectrum and fission spectrum as possible as. In the driven zone, not much neutron multiplication is to be expected, while the hard neutron spectra with average energy about 700 keV is expected. In the active zone the thermal neutron is expected. The assembly will be operated in deep sub-criticality $k_{eff} \approx 0.90 \sim 0.98$ range. The neutron importance φ*, k_{eff}, spatial distribution of neutron flux, neutron spectra and fission rate will be measured for d-T and d-D source respectively.

Intense proton ion source

An electron cyclotron resonance (ECR) ion source is selected for the source of our verification facility system. The microwave power generated by a 2.45 GHz-1 kW magnetron is coupled into the copper chamber (54×72 mm in cross section and 36 mm long) through a three stubs tuning unit and a ridged wave guide. Inside the chamber, a ϕ54 mm in diameter quartz tube which is tightly fixed by a BN disk and a plasma electrode is placed to confine the plasma. A BN plate is placed between the ridged waveguide and plasma chamber to separate the plasma and vacuum. Three holes are made on the waveguide to evacuate the wave guide after the microwave window; with this configuration the gas in the waveguide can be evacuated quickly to avoid interfering the discharge. The microwave window for vacuum sealing is placed behind a bend section in order to avoid any damage due to the back streaming electrons. The microwave system including its power supply is placed on the 75 kV high voltage platform.

A 65 mA hydrogen beam can be routinely extracted from a ϕ6.5 aperture of the source. The emittance of the extracted beam is measured by a multi-slits and single thread emittance-measuring unit. The measured emittance of the total beam at 60 mA, 60 kV, 50 cm downstream of the ion source is 0.129 πmm.mrad. At a specific extraction distance, an adequate extraction voltage always can be found for various beam currents to obtain minimum emittance. The proton ratio is measured by analysing a portion of the beam with a mini-deflection magnet. The result shows that proton fraction is more than 80% which satisfied the requirement of the system. The proton fraction slightly varies with the changes of microwave power but no significant effect is found.

The more stringent request concerning the reliability test has been investigated. There are three breakdowns in the 121 hours test, first two breakdowns occur at first five hours and the last one occurs two hours to the end of the test. All three breakdowns caused by self-protection of the power supply of magnetic coils. The beam is restored in one minute by simply restarting the power supply each time. The longest uninterrupted beam time is 110 hours. A solenoid has been installed 0.6 m downstream of the extraction aperture. The primary result shows that the solenoid works as expected, a through investigation of the solenoid is being conducted.

RFQ accelerator study

The structure of RFQ is a four-vane type and designed to accelerate 50 mA peak current of proton beam with input energy of 80 kV. In preliminary research phase, the 352.2 MHz RF system will be operated in pulse mode. CERN kindly provided IHEP with some RF equipment. Because the given RF system was used for CW operation at CERN before, to apply them to our pulse mode operation, some modifications and improvements are necessary. We have made some indispensable assemblies, and also did some tests and commissioning of every sub-system. At present, we have already finished the 100 kV power supply test and long pulse floating desk hard tube modulator test. Furthermore, the initial high power conditioning of the klystron is carried out, and output power can reach up to 334 kW in CW mode and 402 kW in pulse mode.

The fabrication of the RFQ copper model is being performed in a company in Shanghai, China. At first, some tests for development the mechanical technology have be done, for example, the brazing technology for assembling four vanes together with required mechanical tolerance, the characteristics of melting filler, the structure surface and the vacuum leak; the drilling of the coolant hole through the 1.2 meter RFQ cavity with 12 mm in diameter; the precision machining of the vane electrodes on the numerical controlled mill.

ADS related nuclear data

The new nuclear reaction theoretical models code MEND, which can give all kinds of reaction cross sections and energy spectra for six outgoing light particles (neutron, proton, alpha, deuteron, triton, and helium), gamma and recoil nuclei in the energy range up to 250 MeV, is being developed. The incident particle can be neutron, proton, alpha, deuteron, triton and helium. A programme [7] for automatically searching optimal optical potential parameters in E< 300 MeV energy region has been developed. By this code, the best optical potential parameters can be searched automatically to fit with the relevant experimental data of total cross sections, nonelastic scattering cross sections, elastic scattering cross sections and elastic scattering angular distributions. Nuclear data evaluation method has been developed for ADS. According to the experimental data of neutron-induced reactions, and theoretical model calculation codes UNF [8], ECIS and DWUCK, all cross sections of neutron induced reaction, angular distributions, double differential cross sections for neutron, proton, deuteron, triton, helium and alpha emission, γ-ray production cross sections and γ-ray production energy spectrum are calculated and evaluated at incident neutron energies from 10^{-5} eV to 20 MeV.

Since the recoil effect is taken into account, the energy for whole reaction processes is balanced. Nuclei have been evaluated as follows:

$^{50,52,53,54,nat}Cr$

$^{54,56,57,58,nat}Fe$

$^{90,91,92,94,96,nat}Zr$

$^{112,114,115,116,117,118,119,120,122,124,nat}Sn$ [9]

$^{180,182,183,184,186,nat}W$

$^{204,206,207,208,nat}Pb$

^{209}Bi [10]

^{232}Th [11]

$^{233,234,235,238}U$

By using advanced nuclear models that account for details of nuclear structure and the quantum nature of the nuclear scattering, nuclear data are calculated and evaluated for both incident neutrons and incident protons at incident neutron energy from 20 to 250 MeV as follows:

$^{50,52,53,54}Cr$

$^{54,56,57,58}Fe$

$^{90,91,92,94,96}Zr$

$^{180,182,183,184,186}W$

$^{204,206,207,208}Pb$

and at incident proton energy from threshold energy to 250 MeV as follows:

$^{54,56,57,58}Fe$

$^{180,182,183,184,186}W$

$^{204,206,207,208}Pb$

^{209}Bi [12]

ADS related target physics

The calculations for the standard thick target were made by using different codes. The simulation of the thick Pb target with length of 60 cm, diameter of 20 cm bombarded with 800, 1 000, 1 500 and 2 000 MeV energetic proton beam was carried out. The yields and the spectra of emitted neutron were studied. The spallation target was simulated by SNSP, SHIELD, DCM\CEM (Dubna Cascade Model \Cascade Evaporation Mode), and LAHET codes. The neutron yields calculated by SHIELD and DCM\CEM were in agreement within ±10%.

Material development for ADS beam window

Three heats of 9Cr2WVTa steel have been smelted. The mechanical properties of the smelted 9Cr2WVTa steel have been investigated. It is indicated that the C and Mn content as well as the heat treatment technologies affect the mechanical properties, therefore, the optimum of the elements content and the heat treatment technologies will be the key issues for the improvement of the 9Cr2WVTa steel. This research is being performed at the moment. In order to get the martensitic structure and increasing its mechanical properties, the quenching treatment was performed. It can be seen that the black dots in the matrix become more and more with the increasing of the tempering temperature, this may results from the carbides become more and more with increasing of the temperature. There are little carbides in the matrix without tempering. The measurement results of the micro-hardness indicated that the hardness decreases with the increasing of the tempering temperature, it may results from the dissolution of the martensitic under the increasing of the temperature.

ADS related material radiation effects study

The spallation neutron source system is one of the three key parts of ADS, which provides source neutrons of ~10^{18} n/sec for the burning-up of fuels. It is mainly composed of the target and beam window. Stainless steels and tungsten are important candidate materials of the beam window and the spallation neutron source target. They are irradiated by high-energy and intense protons and neutrons during operation. The accumulated dose could reach a couple of hundred dislocations per atom (‘dpa’) per year, and radiation damage is very severe in them. The radiation damage study of the spallation target and beam window materials is of great importance for the understanding of their lifetimes and the safe operation of the ADS.

Dependence of radiation damage in the modified 316L stainless steel has been investigated on irradiation temperature from room temperature to 802°C at 21 and 33 dpa and on irradiation dose up to 100 dpa at room temperature by the heavy ion irradiation simulation and positron annihilation lifetime techniques. A radiation swelling peak was observed at ~ 580°C where the vacancy cluster contains 14 and 19 vacancies and has an average diameter of 0.68 nm and 0.82 nm, respectively for the 21 and 33 dpa irradiations. The size of the vacancy clusters increases with the increasing of irradiation dose, and the vacancy cluster produced at 100 dpa consists of eight vacancies and reaches a size of 0.55 nm in diameter. The experimental results show that the radiation damage in this modified 316L stainless steel is more sensitive to irradiation temperature than to irradiation dose.

Before this experiment, radiation damage and its detailed thermal annealing behaviour in αAl_2O_3 irradiated at the equivalent dose, respectively, by 5.28×10^{16} cm^{-2} 85 MeV ^{19}F ions and by $3x10^{20}$ cm^{-2} $E_n \geq 1$ MeV neutrons have been investigated by the positron annihilation lifetime technique. The experimental results show that all the positron annihilation parameters of lifetime and intensity in the heavy ion irradiated αAl_2O_3 are in good agreement with the ones in the neutron irradiated αAl_2O_3, and verify that heavy ion irradiation can well simulate neutron (proton) irradiation.

Fundamental research of partitioning

Because of the limited uranium resource in China, we pursue the closed-cycle policy for the nuclear fuels instead of once-through mode. We also believe that reprocessing is essentially a nonproliferation process. Our commercial spent fuel reprocessing plant is anticipated to be built around 2020. So, disposal of the high-level wastes is not an urgent matter in China for the time being. Some preliminary work is under way, such as siting of repository and migration behaviour of some key nuclides under deep geological conditions. Some new progress of partitioning has been reported in the Fifth Joint Workshop between China and Korea on Nuclear Waste Management and Nuclear Fuel Cycle, September 5-8, 2004, Xi'an, China, jointly organised by China Institute of Atomic Energy and Korea Atomic Energy Research Institute. For example, study on elimination of interface crud by using acetohydroxamic acid in separation of HLLW with amido podand [13], reactions of formo-(aceto) hydroxamic acid with Np and Pu and their application in separation of Np and Pu from U [14], the partition of Pu(III, IV) between dilute TBP/OK and aqueous phase and the stability of dimethyl hydroxylamine under high acidity [15], capacity research on adsorption of zirconium by silica gel in nitric acid solution [16] has been reported.

In this Workshop, Professor Gu Zhongmao pointed out some disadvantages of thermal reactor fuel cycles. Taking into account the fact that the build-up of spent fuel in China will not be a big burden in the coming 20 or 30 years, he suggested it would be reasonable for China not to follow the present practice of PWR fuel cycles in other countries and could directly transit from PWR to FR fuel cycles, i.e., the separated Pu from reprocessing of PWR spent fuel would be used to fabricate the MOX fuel and to be fed to FRs, followed by the later FR fuel cycles [17]. In this case, it would be reasonable to build the reprocessing plant by 2030 instead of previously expected 2020. Both aqueous and dry reprocessing processes may be needed to achieve the closed nuclear fuel cycles in China.

Conclusion

For long term and sustainable nuclear energy development, FBR/ADS is an option in fuel circulation and energy generation. The two new nuclear systems FBR and ADS have been started to develop with a rather moderate project in China and they are all still in the early stage. The goal for our FBR/ADS research is to establish the scientific and technological foundation for the future development of the FBR/ADS research step by step.

Acknowledgements

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IAEA ACTIVITIES IN THE AREA OF PARTITIONING AND TRANSMUTATION

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Abstract

In recent years, in various countries and at an international level, more and more studies have been carried out on advanced and innovative waste management strategies (i.e., actinide separation and elimination). In the frame of the Project on *Technology Advances in Fast Reactors and Accelerator Driven Systems for Actinide and Long-lived Fission Product Transmutation* (http://www.iaea.org/inis/aws/fnss/), the IAEA initiated a number of activities on utilisation of plutonium and transmutation of long-lived radioactive waste, accelerator driven systems, thorium fuel options, innovative nuclear reactors and fuel cycles, non-conventional nuclear energy systems, and fusion/fission hybrids. The paper will present an overview of these activities.

Introduction

Based on an experience of more than 10^5 reactor-years, nuclear power is a mature technology that makes a large contribution to the energy supply worldwide. As of end 2004, there were 441 nuclear power plants operating in the world with a total net installed electrical capacity of 367 GW, and 25 nuclear power plants under construction [1]. Nuclear power supplied 16% of global electricity generation in 2002 [2].

According to the projections published by the Intergovernmental Panel on Climate Change (IPCC), the median electricity increase till 2050 will be by a factor of almost 5. It is reasonable to assume that nuclear energy will play a role in meeting this demand growth. However, there are four major challenges facing the long-term development of nuclear energy as a part of the world's energy mix: improvement of the economic competitiveness, meeting increasingly stringent safety requirements, adhering to the criteria of sustainable development, and public acceptability. Meeting the sustainability criteria is the driving force behind the topic of this paper. More specifically, in this context sustainability has two aspects: natural resources and waste management. IAEA's activities in the area of Partitioning and Transmutation (P&T) are mostly in response to the latter. While not involving the large quantities of gaseous products and toxic solid wastes associated with fossil fuels, radioactive waste disposal is today's dominant public acceptance issue. In fact, small waste quantities permit a rigorous confinement strategy, and mined geological disposal is the strategy followed by some countries. Nevertheless, political opposition arguing that this does not yet constitute a safe disposal technology has largely stalled these efforts. One of the primary reasons that are cited is the long life of many of the radioisotopes generated from fission. This concern has led to increased R&D efforts to develop a technology aimed at reducing the amount of long-lived radioactive waste through transmutation in fission reactors or accelerator driven hybrids. In recent years, in various countries and at an international level, more and more studies have been carried out on advanced and innovative waste management strategies (i.e., actinide separation and elimination). In the frame of the Project on *Technology Advances in Fast Reactors and Accelerator Driven Systems for Actinide and Long-lived Fission Product Transmutation* (http://www.iaea.org/inis/aws/fnss/), the IAEA initiated a number of activities on utilisation of plutonium and transmutation of long-lived radioactive waste, accelerator driven systems, thorium fuel options, innovative nuclear reactors and fuel cycles, non-conventional nuclear energy systems, and fusion/fission hybrids. The paper will present an overview of these activities.

IAEA Activities

As in all the other fields of advanced nuclear power technology development, the Agency is relying also in the P&T area on broad, in-depth staff experience and perspective. The framework for all the IAEA activities in the P&T area is the Technical Working Group on Fast Reactors (TWG-FR). In responding to strong common R&D needs in the Member States, the TWG-FR acts as a catalyst for international information exchange and collaborative R&D.

Given the common technical ground between plutonium utilisation R&D activities and the development of technologies for the transmutation and utilisation of long-lived fission products and actinides, both activities are performed within the framework of a single Agency project: *Technology Advances in Fast Reactors and Accelerator Driven Systems for Actinide and Long-lived Fission Product Transmutation* [3].

The TWG-FR is a standing working group within the framework of the IAEA. It provides a forum for exchange of non-commercial scientific and technical information, and a forum for international co-operation on generic research and development programmes on advances in fast reactors and fast spectrum accelerator driven systems. Its present members are the following 14 IAEA Member States: Belarus, Brazil, China, France, Germany, India, Italy, Japan, Kazakhstan, Republic of Korea, Russia, Switzerland, United Kingdom, and United States of America, as well as the OECD/NEA, and the EU (EC). The TWG-FR advises the Deputy Director General-Nuclear Energy on status of and recent results achieved in the national technology development programmes relevant to the TWG-FR's scope, and recommends activities to the Agency that are beneficial for these national programmes. It furthermore assists in the implementation of corresponding Agency activities, and ensures that through continuous consultations with officially nominated representatives of Member States all the project's technical activities performed within the framework of the Nuclear Power Technology Development sub-programme are in line with expressed needs from Member States.

The scope of the TWG-FR is broad, covering all technical aspects of fast reactors and ADS research and development, design, deployment, operation, and decommissioning. It includes, in particular: design and technologies for current and advanced fast reactors and ADS; economics, performance and safety of fast reactors and ADS; associated advanced fuel cycles and fuel options for the utilisation and transmutation of actinides and long-lived fission products, including the utilisation of thorium. Given the TWG-FR's broad scope, the coverage will generally be in an integrative sense to ensure that all key technology areas are covered. Many specific technologies are addressed in detail by other projects within the IAEA and in other international organisations. The TWG-FR keeps abreast of such work, avoiding unproductive overlap, and engages in co-operative activities with other projects where appropriate. The TWG-FR thus coordinates its activities in interfacing areas with other Agency projects, especially those of the International Working Group on Nuclear Fuel Cycle Options, and the Department of Nuclear Safety, as well as with related activities of other international organisations (OECD/NEA, and EC).

Recent accomplishments

In responding to Member States' needs for information exchange in the fields covered by the Project's scope, the IAEA has published a series of Technical Reports (IAEA-TECDOCs) dealing with innovative reactor technology development in view of the utilisation and transmutation of actinides and long-lived fission products.

Justified by the existing and growing interest in many IAEA Member States to investigate the potential of advanced thorium fuel cycles and the related reactor technologies, the first Technical Report [4] attempts an assessment of the advantages, shortcomings, and options of the thorium fuel under current conditions, with the aim of identifying new research areas and fields of possible co-operation within the framework of the IAEA's advanced technology development projects. Apart from current commercial reactors, the report covers all types of evolutionary and innovative nuclear reactors, including molten salt reactors and hybrid systems. The report addresses the main physics aspects of thorium fuelled reactor cores, assesses advantages and disadvantages of thorium fuel utilisation, presents the various options and concepts under investigation, and reviews remaining problems and uncertainties linked to thorium fuel utilisation and reactor technologies based on thorium fuel. Two issues are identified as main reasons for the renewed interest in thorium fuel cycles: the potential to incinerate plutonium and reduce actinide production, on the one side, and better material properties and fuel behaviour, on the other side.

The report's most important conclusion is that there is a need for a unified systematic approach in assessing thorium fuel utilisation: a methodology (metrics) to evaluate the performance parameters of the thorium fuel cycle must be developed. This methodology would have to define the performance parameters matrix as well as the algorithms for the evaluation. Another important conclusion highlighted in the report is the necessity to develop and maintain a database of all available information relevant to thorium fuel cycles, their utilisation and related reactor technology.

To further investigate one of the two issues identified as main reasons for the renewed interest in thorium fuel cycles, i.e., their potential to incinerate plutonium and reduce actinide production, the IAEA has implemented a Coordinated Research Project (CRP) on *Potential of Thorium Based Fuel Cycles to Constrain Plutonium and to Reduce Long Lived Waste Toxicity*, and published the final results as an IAEA-TECDOC [5]. This CRP examined through computer simulations the different fuel cycle options in which plutonium can be recycled with thorium to incinerate plutonium. In the course of the CRP, the participants performed three benchmark tasks for different reactor concepts. Their incentive was the comparison of the various codes and nuclear data. The assessment of thorium fuelled thermal reactors in view of their potential for the incineration of plutonium and of a possible combined reduction of the waste radio-toxicity has been performed. Generally, the agreement of the benchmark results was very satisfying. The participants concluded that the results obtained constitute a sufficiently reliable basis for overall conclusions on the potential of thorium-based cycles to constrain plutonium and to reduce the long-term potential radiotoxic hazard of the waste. The overall conclusions can be summarised as follows: Generally, there is a remarkable potential to effectively constrain the production of plutonium and to reduce existing plutonium stockpiles by implementing the thorium fuel cycle in a large number of current reactors. This path offers a promising near-future plutonium management option. However, plutonium incineration in thermal reactors turns out to be less effective from the point of view of the reduction of the long-term radio-toxicity of the nuclear waste. A reduction by an order of magnitude or more of the potential long term radiotoxic hazard of the waste seems not to be achievable by any of the considered plutonium incinerating thermal reactors. Most of the calculations performed for LWR plutonium indicate that the waste radio-toxicity will be decreased by not more than a factor of 2 to 4, and only for an intermediate period; the waste radio-toxicity is even increased during the first decades and for extremely long times after disposal.

In another Technical Report [6], the IAEA reviewed the major R&D developments in the area of lead and lead–bismuth eutectic cooled reactor technology, for both critical and sub-critical systems. Particular emphasis is put on reviewing critical and sub-critical concepts, coolant properties, and experimental and analytical validation work. The fast reactor concept BREST-OD-300 under investigation in Russia, as well as various conceptual designs of heavy liquid metal cooled fast reactors pursued in Japan are described. Research and development work on hybrid (accelerator driven) sub-critical systems ongoing in various Member States are reviewed. The report concludes that nuclear energy is a realistic solution to satisfy the energy demand, considering the limited resources of fossil fuel, its uneven distribution in the world and the impact of its use on the planet, as well as the expected doubling of the world population in the 21st century and tripling of the electricity demand (especially in the developing countries). The report stresses that the development of innovative nuclear technologies must be pursued meeting the following requirements: (a) deterministic exclusion of any severe accident; (b) proliferation resistance; (c) cost competitiveness with alternative energy sources; (d) sustainable fuel supply; and (e) innovative solutions to the radioactive waste management problem.

Potential advantages of accelerator driven systems — apart from their intrinsic low production of long lived radioactive waste, and transmutation capability — are also enhanced safety characteristics and better long term resources utilisation (e.g. in connection with thorium fuels). The Technical Reports [7, 8] review the R&D programmes that are being undertaken by various institutions in IAEA Member States to substantiate these claims and advance the basic knowledge in this innovative area of nuclear energy development, and examine needs and possible opportunities for international collaboration. While long term objectives for developing innovative nuclear systems for energy production and transmutation may not be unanimously agreed upon by the different groups participating in this effort, it is clear that in many cases the short term goals are similar. Therefore, quite a few generic R&D areas that would benefit from international collaboration are identified. The most important technical issues identified and discussed are: (a) Thermal fatigue due to beam trips; (b) Toxicity of the spallation products; (c) The lack of a safety strategy for severe accidents with fertile-free transuranics fuel; and (d) The lack of data on irradiation damage effects on the structural properties of the beam window and the adjacent core, which are induced by both proton and neutron irradiation. The reports conclude on the need and opportunity for collaboration in the following areas: (a) Major demonstration facilities, for which international participation should be considered; (b) Testing of special effects (e.g. fuels and materials tests, and zero power coupled systems) which offers practical opportunities for dividing up the work, and (c) Analytical benchmarks.

Ongoing and planned activities

In response to Member States information exchange needs, the project on *Technology Advances in Fast Reactors and Accelerator Driven Systems for Actinide and Long-lived Fission Product Transmutation* is preparing a series of publications on R&D topics of interest, specifically, Technical Reports (a) to review solid and mobile fuels for partitioning and transmutation systems, (b) on theoretical and experimental studies of heavy liquid metal thermal hydraulics, (c) to perform a comparative assessment of the dynamics and safety characteristics of transmutation systems, (d) to update the status of accelerator driven systems research and technology development, and (e) on the use of fusion / fission / accelerator based systems for the utilisation and transmutation of actinides and long-lived fission products. Another activity planned for 2005 addresses formation and training needs expressed by the Member States: in collaboration with the International Centre for Theoretical Physics (ICTP), the IAEA is organising the Workshop on *Technology and Applications of Accelerator Driven Systems*, in Trieste, Italy, from 17-28 October 2005. The Workshop will consist of lectures, tutorials, and computer exercises covering all the areas of ADS research and technology development, as well as the applications, i.e., accelerator technology, nuclear data, ADS concepts (design), simulation methods, ADS safety, and fuel cycle issues.

With regard to collaborative R&D, the project has an ongoing (2002-2006) CRP on Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste, and will start a new CRP (2005-2009) on Analytical and Experimental Benchmark Analyses of Accelerator Driven Systems (ADS). The former CRP was joined by participants from 17 institutions in 13 Member States, and the EC (JRC). Its objective is to produce a comparative assessment of the transient behaviour of advanced transmutation systems, both critical and sub-critical. The CRP performs benchmarks on critical liquid metal, and gas cooled fast reactor, heavy liquid metal, and gas cooled ADS, critical and sub-critical molten salt concepts, and fusion-fission hybrid sub-critical systems. The objective of the latter CRP is to improve the understanding of the physics of the coupling of external neutron sources with sub-critical cores. Experimental backing of analytical benchmarks is the major thrust of this CRP, and the participants will apply integrated calculation schemes to perform computational and experimental benchmark analyses.

Last but not least, the Agency has implemented the "ADS Research and Development Database". It provides information about ADS related R&D programmes, existing and planned experimental facilities as well as programmes, methods and data development efforts, design studies, and so forth. While operational on the WWW and open to all users (*http://www-adsdb.iaea.org/index.cfm*), the database has to rely on content contributed by the interested community. Data and information can be provided on-line, and contributions are solicited (the author will gladly provide, upon request, access privileges as editor to everybody wanting to contribute content).

Conclusions

For nuclear energy to remain a long-term option in the world's energy mix, nuclear power technology development must meet sustainability goals with regard to fissile resources and waste management. The utilisation of breeding to secure long-term fuel supply remains the ultimate goal of fast neutron spectrum system. Plutonium recycling in fast reactors, as well as incineration/ transmutation of minor actinides and long-lived fission products in various hybrid reactor systems (e.g., ADS) offers promising waste management options. Several R&D programmes in various Member States are actively pursuing these options, along with the energy production and breeding mission of fast reactor systems.

In line with the statutory objective expressed in Article II (The Agency shall seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world. It shall ensure, insofar as it is able, that assistance provided by it or at its request or under its supervision or control is not used in such a way as to further any military purpose), the IAEA will continue to assist the Member States' activities, also in the area of advanced technology development for utilisation and transmutation of actinides and long-lived fission products, by providing an umbrella for information exchange and collaborative R&D to pool resources and expertise.

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LIST OF RELATED IAEA PUBLICATIONS

(Most of these publications can be downloaded as pdf files from the Web Site of the project on *Technology Advances in Fast Reactors and Accelerator Driven Systems for Actinide and Long-lived Fission Product Transmutation*: http://www.iaea.org/inis/aws/fnss/)

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PARTITIONING AND TRANSMUTATION RESEARCH IN THE EURATOM FIFTH AND SIXTH FRAMEWORK PROGRAMMES

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Abstract

Partitioning and Transmutation (P&T) of long-lived radionuclides in nuclear waste is one of the most notable research areas of the EURATOM Fifth (1998-2002) as well as the sixth (2002-2006) Framework Programmes (FP). The objective of research work in this area is to provide a basis for evaluating the practicability of P&T, on an industrial scale, for reducing the amount of long lived radionuclides to be disposed of thus easing the waste management problem. In FP5, there are 15 projects in the area of P&T with a total budget of about 69 M€ of which the EU contribution is about 28 M€. A network ADOPT co-ordinates the activities of the accelerator driven system (ADS) design project with those of the four clusters, one on chemical separation (i) PARTITION and three on transmutation, (i) Basic studies (BASTRA), (ii) Technological studies (TESTRA) and (iii) Fuel studies (FUETRA). Each of these clusters is formed by 3-4 projects, which are briefly described.

Eleven projects have been completed and the remaining ones are due to be completed at the latest in about 2 years. The FP6 is geared towards creating "European Research Area" (ERA) by strongly increasing the collaborative research and innovation efforts across Europe. An element in achieving ERA is the organisation of siseable European "Networks of Excellence" and "Integrated Projects". The EC budget of the P&T research in FP6 is only marginally higher but the emphasis is on organising fewer but fairly large coherent projects. There are two Integrated Projects, one on partitioning "EUROPART" (started in 2004) and the other on transmutation "EUROTRANS" (as yet under negotiations) and there is a Specific Targeted Research Project on the impact of P&T on waste management "Red-IMPACT" (started in 2004). The total funding of these three projects, of 3-4 year duration, is approximately 56 M€ whereas the EC contribution is about 31 M€. A brief outline of these projects is given. A further call for proposals is planned in Spring 2005. International co-operation in the area of P&T with non-EU countries including the Commonwealth of Independent States (CIS) is also outlined.

Introduction

The priorities for the European Union's research and development activities for the period 1998-2002 are set out in the Fifth Framework Programme (FP5) [1]. FP5 focuses on a limited number of research areas combining technological, industrial, economic, social and cultural aspects. FP5 and its predecessors have contributed effectively to the policy of supporting science and technology by encouraging co-operation between research players of the Member States. Despite this achievement, no specific European research policy was seen to emerge. National research programmes were still undertaken to a large extent independently of one another.

One of the objectives of EU is to achieve greater co-operation between Member States' research strategies and a mutual opening up of programmes. With the challenges and prospects opened up by the technologies of the future, there is a need that European research efforts and capacities should be more thoroughly integrated. With this view in mind, the European Commission launched the so-called "European Research Area" (ERA) initiative in January 2000 [2]. The Sixth Framework Programme (FP6) [3] encompassing the period 2002-2006 is geared to make ERA a reality [4].

The overall organisation of FP6 reflects the broad avenues of approach that are implicit in the proposed implementation of ERA. FP6 has three main blocks of activities:

- Integrating research in the well focussed research priority areas principally by using new research implementation instruments such as Networks of Excellence (NoE) and Integrated Projects (IP).

- Structuring the ERA by research and innovation, human resources and researcher mobility, research infrastructure and science and society issues.

- Strengthening the foundations of ERA by networking of national research and opening up of national programmes, closer links between EU and other European organisations (such as CERN), benchmarking of research policies, mapping of excellence etc.

In this context, the scientific and technical goals of the EURATOM FP6 specific programme "Research and Training Programme on Nuclear Energy" is to help exploit the full potential of nuclear energy, both in the long and short term. Its development and exploitation is to be done in a sustainable manner while combating the climate change and reducing the energy dependency of the EU. Research and development activities in this programme have been subdivided into (a) Controlled thermonuclear fusion, (b) Management of radioactive waste, (c) Radiation protection and (d) Other activities in the field of nuclear technologies and safety.

Controlled thermonuclear fusion is perceived to be one of the long-term options for energy supply whereas nuclear fission presently provides about 35% of the EU's electrical power. Some of the fission power plants of the current generation will continue to operate for at least 20 years. In the short term, the priority is to find a more permanent and safe solution for the management of long-lived, high-level waste that is acceptable to society. A priority in this area is to establish a sound technical basis for demonstrating the safety of disposal of spent fuel and long-lived radioactive wastes in geological repositories. This is to be supported by evaluating the practicability, on an industrial scale, for reducing the amount and/or hazard of the waste to be disposed of by partitioning (chemical separation) and transmutation (nuclide conversion). This is further supplemented by exploring the potential of system concepts that would by themselves produce less waste in nuclear energy generation.

The EURATOM Fifth Framework Programme (FP5) (1998-2002)

The Fifth Framework Programme of the European Atomic Energy Community (EURATOM) has two specific programmes on nuclear energy, one for indirect research and training actions managed by the Research Directorate General (DG RTD) and the other for direct actions under the responsibility of the Joint Research Centre (JRC) of the European Commission (EC). The strategic goal of the first one, "Research and training programme in the field of nuclear energy," is to help exploit the full potential of nuclear energy in a sustainable manner, by making current technologies even safer and more economical and by exploring promising new concepts [1]. This programme includes a key action on controlled thermonuclear fusion, a key action on nuclear fission, research and technological development (RTD) activities of a generic nature on radiological sciences, support for research infrastructure, training and accompanying measures. The key action on nuclear fission and the RTD activities of a generic nature are being implemented through indirect actions, i.e. research co-sponsored and co-ordinated by DG RTD, but carried out by external public and private organisations as multi-partner projects. The total FP5 budget available for these indirect actions is 193 M€.

The key action on nuclear fission comprises four areas:

(i) operational safety of existing installations;

(ii) safety of the fuel cycle;

(iii) safety and efficiency of future systems; and

(iv) radiation protection.

P&T activities lie in the area of the safety of the fuel cycle which also encompasses waste and spent fuel management and disposal. The implementation of the key action on nuclear fission is made through targeted calls for proposals with fixed deadlines. Following the three calls for proposals made since the start of FP5, 15 projects were funded in the area of P&T, with a total budget of 69 M€ out of which EU contribution is 28 M€.

The research activities on P&T in the EURATOM Fifth Framework Programme

The objective of the research work carried out under FP5 is to provide a basis for evaluating the practicability of partitioning and transmutation, on an industrial scale, for reducing the amount of long lived radionuclides to be disposed of. The work on partitioning concerns the experimental investigation of efficient hydro-metallurgical and pyrochemical processes for the chemical separation of long-lived radionuclides from high-level liquid waste. The work on transmutation is related to the preliminary design studies of an accelerator driven sub-critical system (ADS) and acquisition of basic and technological data necessary for its development including the development of fuel and targets for an ADS [5].

The selected projects in this area address various scientific and technical aspects of P&T and have therefore been regrouped. A network ADOPT co-ordinates the activities of the accelerator driven system (ADS) design project with those of the four clusters of FP5 projects in the area of P&T (see Figure 1). One cluster is on chemical separation of radionuclides **(PARTITION)** and there are three on transmutation: (i) Basic studies **(BASTRA),** (ii) Technological studies **(TESTRA)** and (iii) Fuel studies **(FUETRA).**

ADOPT network

The objectives of ADOPT network (see Table 1) are:

(i) to formulate actions with a view to promote consistency between FP5 funded projects and national programmes;

(ii) to review overall results of the FP5 projects;

(iii) to identify gaps in the overall programme of P&T research in Europe;

(iv) to provide input to future research proposals and guidelines for R&D orientation; and

(v) to maintain relations with international organisations and countries outside the EU involved in P&T and ADS development.

Table 1. **Advanced options for P&T (ADOPT) network and preliminary design studies for an experimental ADS (PDS-XADS)**

Acronym	**Subject of research**	**Co-ordinator** (country)	**No. of partners**	**Start date & duration**	**EC funding** (M€)
ADOPT Network	Thematic Network on Advanced Options for P&T	SCK/CEN (B)	16	01-11-01 36 m	0.4
PDS-XADS	Preliminary Design studies of an experimental accelerator driven system	Framatome-ANP (F)	25	01-11-01 36 m	6.0

Design studies of an experimental ADS

Successful operation of an ADS together with the coupling of an accelerator to the neutron spallation target and the sub-critical core is a first step for demonstrating the practicability of this type of transmuter on an industrial scale. Aim of the PDS-XADS project (see Table 1) is to make well documented study with supporting evidence to choose and adopt the most promising technical concepts for ADS. It also addresses the critical points of the entire system, identifying the research and development (R&D) required in support, the definition of the safety and licensing issues, assessing the preliminary cost of the installation and consolidating the road mapping of the XADS development. The assessment and comparison studies of the different conceptual designs of the main systems (accelerator, spallation target unit, sub-critical core, primary system) has allowed to identify the most promising solution(s) (Pb-Bi cooled system with a gas back-up) which would be studied in detail during the next phase of the design activities.

Partitioning Projects

The PARTITION cluster includes three projects, the main characteristics of which are given in Table 2. The first one, **PYROREP**, aims at assessing flow sheets for pyrometallurgical processing of spent fuels and targets. Two methods, salt/metal extraction and electrorefining, investigate the possibility of separating actinides from lanthanides. Materials compatible with corrosive media at high temperature are selected and tested. Electrochemical studies of Ln and An have provided basic data to assess electrochemical separation methods in molten chloride salts. New pyrometallurgic equipments have also been developed.

Table 2. **PARTITIONING cluster projects**

Acronym	**Subject of research**	**Co-ordinator** (country)	**No. of partners**	**Start date & duration**	**EC funding** (M€)
PYROREP	Pyrometallurgical Processing Research	CEA (F)	7	01-09-00 36 m	1.5
PARTNEW	Solvent Extraction Processes for Minor Actinides (MA)	CEA (F)	10	01-09-00 36 m	2.2
CALIXPART	Selective Extraction of MA by Organised Matrices	CEA (F)	9	01-10-00 40 m	1.4

The two other projects deal with the development of solvent extraction processes to separate minor actinides (americium and curium) from high-level liquid waste (HLLW). In the project **PARTNEW**, the minor actinides are extracted in two steps. They are first co-extracted with the lanthanides from HLLW (by DIAMEX process), then separated from the lanthanides (by SANEX process). The DIAMEX process, based on the use of the malonamide DMDOHEMA is mature. The SANEX process based on BTP, while giving good An(III)/Ln(III) separation performances does not appear suitable for an industrial development owing to the insufficient stability of the BTP extractant.

The **CALIXPART** project deals with the synthesis of more innovative extractants. Functionalised organic compounds, such as calixarenes, are synthesised with the aim of achieving the direct extraction of minor actinides from HLLW. The extraction capabilities of the new compounds are studied together with their stability under irradiation. About 160 extractants for removal of minor actinides from high activity liquid wastes were synthesised. The most promising molecules able to separate minor actinides from lanthanides were tested on real wastes from the PUREX process.

Transmutation projects

(i) BASTRA cluster

Three projects are grouped in the cluster of basic studies on transmutation (BASTRA) (see Table 3). The **MUSE** project aims to provide validated analytical tools for sub-critical neutronics, data and a reference calculation tool for ADS study. The experiments are carried out by coupling a pulsed D-T/D-D neutron generator source (GENEPI) to the MASURCA facility loaded with MOX fuel operated as a sub-critical system with different coolants (such as sodium and lead). Cross-comparison of codes and data has been done.

The other two projects deal with nuclear data. The objective of the **HINDAS** project is to collect most of the nuclear data necessary for ADS applications. This is achieved by basic cross-section measurements at different European accelerator facilities, nuclear model simulations and data evaluations in the 20-200 MeV energy region and beyond. Iron and lead (materials used for ADS) and uranium have been chosen to have a representative coverage of the periodic table.

The **n-TOF-ND-ADS** project aims at the production, evaluation and dissemination of neutron cross sections for most of the radioisotopes (actinides and long-lived fission products) considered for transmutation in the energy range from 1 eV up to 250 MeV. Measurements have been carried out at the n-TOF facility at CERN, at the GELINA facility in Geel and using other neutron sources located at different EU laboratories.

Table 3. **Basic studies for transmutation (BASTRA) cluster projects**

Acronym	**Subject of research**	**Co-ordinator** (country)	**No. of partners**	**Start date & duration**	**EC funding** (M€)
MUSE	Experiments for sub-critical neutronics validation	CEA (F)	13	01-10-00 49m	2.0
HINDAS	High and intermediate energy nuclear data for ADS	UCL (B)	16	01-09-00 39m	2.1
n-TOF-ND-ADS	ADS nuclear data using time-of-flight facility	CERN(CH)	18	01-11-00 50m	2.4

(ii) TESTRA cluster

Four projects are grouped in the cluster of technological studies on transmutation (TESTRA) (see Table 4). This cluster deals with the investigation of radiation damage induced by products of spallation reactions in materials, of the corrosion of structural materials by lead alloys and of fuels and targets for actinide incineration.

The **SPIRE** project addresses the irradiation effects on an ADS spallation target. The effects of spallation products on the mechanical properties and microstructure of selected structural steels (e.g. martensitic steels) have been investigated by ion beam irradiation and neutron irradiation in reactors (HFR in Petten, BR2 in Mol and BOR60 in Dimitrovgrad). Data representative of mixed proton/neutron irradiation have been obtained from the analysis of the SINQ spallation target at the Paul Scherrer Institute in Villigen (CH).

The objective of **TECLA** project is to assess the use of lead alloys both as a spallation target and as a coolant for an ADS. Three main topics are addressed: corrosion of structural materials by lead alloys, protection of structural materials and physico-chemistry and technology of liquid lead alloys. A preliminary assessment of the combined effects of proton/neutron irradiation and liquid metal corrosion has been carried out. Thermal-hydraulic experiments have been performed together with numerical computational tool development.

Table 4. **Technological studies for transmutation (TESTRA) cluster**

Acronym	**Subject of research**	**Co-ordinator** (country)	**No. of partners**	**Start date & duration**	**EC funding** (M€)
SPIRE	Effects of Neutron and Proton Irradiation in Steels	CEA (F)	10	01-08-00 48 m	2.3
TECLA	Materials and Thermal-hydraulics for Lead Alloys	ENEA (I)	16	01-09-00 39 m	2.5
MEGAPIE-TEST	A megawatt heavy liquid metal spallation target experiment with proton beam	FZK (D)	17	01-11-01 61 m	2.4
ASCHLIM	Computational Fluid Dynamics Codes for Heavy Liquid Metals	SCK/CEN (B)	14	01-01-02 12 m	0.12

The major objective of the **MEGAPIE-TEST** Project is to develop and validate the design and operation of a heavy liquid metal (Pb-Bi) spallation target at a level of a megawatt. The project aims to provide a comprehensive database from single-effect experiments, a full-scale thermal-hydraulic simulation experiment, and the first beam-on experiments. In parallel, numerical computational tools will be validated for Pb-Bi target design. The studies include neutronic calculations, materials, corrosion, thermal-hydraulics, structure mechanics, liquid metal technology, safety and licensing issues. Prospects on the extrapolation and applicability of the obtained results to an ADS spallation target will also be given.

The ASsessment of Computational fluid dynamics codes for Heavy LIquid Metals (**ASCHLIM**) project aims at bringing together various actors (industry, research institutions and university) in the field of heavy liquid metals both in the experimental and numerical fields and creating an international collaboration to (i) make an assessment of the main technological problems in the fields of turbulence, free surface and bubbly flow and (ii) co-ordinate future research activities in this area. The assessment is being made on the basis of existing experiments whose basic physical phenomena are analysed through the execution of calculational benchmarks using commercial and research codes.

(iii) FUETRA cluster

There are three projects in this cluster (see Table 5). The objectives of the **CONFIRM** project are to develop methods for fabrication (such as carbo-thermic reduction process) of uranium-free nitride fuels (Pu,Zr)N and to model and test their performance under irradiation up to 20% burn-up in a material test reactor. Carbo-thermic process is also used for the production of (Am, Zr)N pellets at ITU, Karlsruhe. Successful high temperature (≈2500°C) stability tests of (U,Zr)N have been made and a study of C-14 production has been completed.

The objective of the project **THORIUM CYCLE** is to investigate the irradiation behaviour of thorium/plutonium (Th/Pu) fuel at high burn-up and to perform full core calculations for thorium-based fuel with a view to supplying key data related to plutonium and minor actinide burning. Two irradiation experiments are being carried out: (i) four targets of oxide fuel (Th/Pu, uranium/plutonium, uranium and thorium) have been fabricated, irradiated in HFR in Petten and characterised after irradiation, (ii) one Th/Pu oxide target is also irradiated in KWO reactor at Obregheim (D).

The main objective of the **FUTURE** project is to study the feasibility of oxide compounds (Pu, Am)O_2, (Th, Pu, Am)O_2 and (Pu, Am, Zr)O_2 to be irradiated as homogeneous fuel for an ADS. The R&D programme is largely devoted to the synthesis of the compounds, their characterisation (thermal and chemical properties at relevant temperatures) and the development of fabrication processes. Modelling codes will be developed to calculate the fuel performance. The input data for the codes will be based on experimental results. Assessment of the fuel behaviour under accident conditions will be analysed using the experimental data obtained at high temperatures.

Table 5. **Fuel studies for transmutation (FUETRA) cluster**

Acronym	**Subject of research**	**Co-ordinator** (country)	**No. of partners**	**Start date & duration**	**EC funding** (M€)
CONFIRM	Uranium-free nitride fuel irradiation and Modelling	KTH (S)	7	01-09-00 64 m	1.0
THORIUM CYCLE	Development of thorium cycle for PWR and ADS	NRG (NL)	7	01-10-00 66m	1.2
FUTURE	Development of transuranic oxide fuels for transmutation	CEA (F)	7	01-12-01 60 m	1.7

The EURATOM Sixth Framework Programme (FP6) (2002-2006)

Research and development activities of the EURATOM FP6 specific programme “Research and Training Programme on Nuclear Energy” have been subdivided into four areas (a) Controlled thermonuclear fusion, (b) Management of radioactive waste, (c) Radiation protection and (d) Other activities in the field of nuclear technologies and safety.

In the area (b), the priority is to find a permanent and safe solution for the management of long-lived, high-level waste that is acceptable to society. This includes establishing a sound technical basis for the demonstration of long lived high level waste disposal in geological formations. This is to be supported by studies on P&T and further supplemented by exploring the potential of system concepts that would by themselves produce less waste in nuclear energy generation. Combating the decline in both student numbers and teaching establishments by a better integration of European education and training in nuclear safety and radiation protection is another important aim.

The detailed work programme of EURATOM FP6 has been adopted by the EC [6]. In P&T, the research areas include a fundamental assessment of the system and safety aspects of the overall concept of P&T and, in particular, of its impact on waste management and geological disposal. In the area of partitioning, continued R&D of hydrometallurgical and pyrochemical processes is envisaged with a view to the demonstration of the most promising techniques. In the area of transmutation, the development of basic knowledge and technologies for transmutation and evaluation of their industrial practicability, in particular, of transmutation devices such as accelerator driven sub-critical systems (ADS) is proposed [7].

Two Calls for proposals have been made in December 2002 and November 2003 respectively, and a third call is expected to be made in Spring of 2005. In the first two calls, the so-called new instruments (such as Integrated Projects) are used as a priority. The Integrated Projects (IP) are designed to give increased impetus to the Community's competitiveness or to address major societal needs by mobilising a critical mass of research and technological development resources and competencies. Avoiding the micro management, increased autonomy is given to the consortia in the management (both scientific and financial) of projects that will be judged on the global end-results. Specific Targeted Research Projects (STREPS) are sharply focused on research and technological development designed to gain new knowledge either to improve or develop new products, processes or services or to meet other needs of society and Community policies.

The research activities on P&T in the EURATOM Sixth Framework Programme

The following projects in the area of P&T have been selected for funding until now:

(i) RED-IMPACT project [8]

Partitioning, transmutation and conditioning (P&T/C) and waste reduction technologies are expected to reduce the burden associated with radioactive waste management and disposal. P&T is likely to ease the final repository requirements and it will also contribute to the sustainability of nuclear energy in those countries that pursue this source of energy.

The objectives of this 3-year RED-IMPACT project (Total budget 3.5 M€ including EC contribution of 2 M€) are: (i) Assess the impact of P&T on geological disposal and waste management, (ii) Assess economic, environmental and societal costs/benefits of P&T (iii) Disseminate results of the study to stakeholders (scientific, general public and decision makers) and get feedback during the course of the study and (iv) Iterate and refine the work based on stake-holders' feedback to achieve full impact of this study on the implementation of the waste management policy of the European Community.

(ii) EUROPART project [9]

The main objectives of research work in this 3-year project (total budget: 10.3 M€ and EC contribution: 6 M€) are (i) the development of methods for the separation of individual minor actinides that are contained in aqueous nuclear wastes issuing from the reprocessing of uranium oxide (UOX) or mixed oxide (MOX) nuclear spent fuel and (ii) partitioning of all actinides (An) together for recycling e.g. in an Accelerator Driven System following double-strata advanced fuel cycle concept.

Partitioning techniques used are: (i) hydrometallurgy and (ii) pyrometallurgy. In hydro-metallurgy, the partitioning methods are mainly based on the use of solvent extraction methods or extraction by chromatographic methods which will be applied for (a) individual separation of the trivalent Am/Cm/Bk/Cf ions (b) joint partitioning of An and (c) reprocessing of innovative nuclear spent fuels. In pyrometallurgy, the nuclear wastes issuing from the reprocessing of present or future nuclear spent fuels can be dissolved into molten halide salts at temperatures of several hundreds of degrees followed by the separation of individual MAs (from U to Cf) or all actinides. This will be considered by several methods, such as (a) electro-deposition as metals, (b) liquid extraction using a molten metallic solvent and (c) selective precipitation as oxides. The basic properties of An in molten halides and the partitioning processes of An from spent fuel and advanced dedicated fuel cycles will be investigated. The flow-sheet of various processes including the conditioning methods for the wastes to be generated by the partitioning processes will also be established.

Processes for possible industrialisation of partitioning strategies will also be defined. Training and education of the young researchers also constitutes an important part of the work in the project.

(iii) EUROTRANS project [10]

The objective of this 4-year IP EUROTRANS (total budget of 42.3 M€ including 23 M€ of EC contribution) is to carry out a preliminary detailed design of a ≈100 MW experimental facility (realisation in a short-term, say about 10 years) demonstrating the technical feasibility of transmutation in an accelerator driven System (XT-ADS) as well as to accomplish a reference conceptual design (several 100 MW) of a modular generic European Transmutation Demonstrator (ETD) in the long-term.

Subject to negotiations, the experimental facility TRADE-PLUS will be operated, at a level of 100 kW, as a sub-critical device driven by an accelerator (a 40 kW, 140MeV proton cyclotron with a proposed solid tantalum spallation target. U-free oxide fuels such as (Pu, MA, Zr)O2 or CERCER (Pu, MA)O2+MgO or CERMET (Pu, MA)O2 + Mo will be developed with a view to their use both in XT-ADS and ETD and will be qualified in HFR and Phénix reactors. A further assessment of structural materials and heavy-liquid metal (HLM) (Pb-Bi) technologies for transmutation systems both as a spallation target material and coolant will be made. Further development of nuclear data evaluated files and models involving sensitivity analysis and validation of simulation tools will be made.

The EUROTRANS project has assembled a consortium of partners incorporating the most relevant actors in this field and it has a very broad multi-disciplinary scientific, educational and industrial background of partners from countries across Europe as well as three institutes of JRC. The universities across Europe are well represented. To provide education and training (E&T) in the nuclear field to young researchers is an important goal of this project. About 5% of the budget in each domain is assigned to PhD students whereas an additional sum is reserved exclusively for E&T courses.

The outcome of this project is expected to provide a fairly reliable basis for an assessment of the technical feasibility of transmutation by ADS and a first estimate of the cost of an ADS based transmutation system. It is also expected to provide certain important input elements to authorities to decide whether to embark on the detailed engineering design of an ADS for transmutation and its eventual construction.

ADS related research activities in the framework of the International Science and Technology Centre (ISTC)

The International Science and Technology Centre (ISTC) was established by an international agreement in November 1992 as a non-proliferation programme through science co-operation. It is an intergovernmental organisation grouping the European Union, Japan, the USA, Canada, Norway, the Republic of Korea, which are the funding parties, and some countries of the Commonwealth of Independent States (CIS): the Russian Federation, Armenia, Belarus, Georgia, Kazakhstan and Kyrgyzstan. A similar organisation, the Science and Technology Centre in Ukraine (STCU) has been established in 1995, in which the EU, Canada, the USA, Georgia and Uzbekistan are involved.

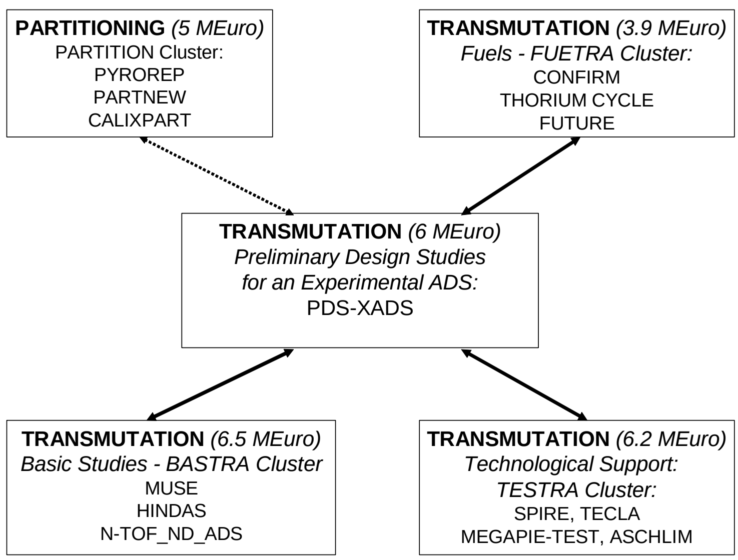
Five topics have been identified by the ISTC Contact Expert Group (CEG) for the ADS related projects: (i) accelerator technology, (ii) basic nuclear and material data and neutronics of ADS, (iii) targets and materials, (iv) fuels related to ADS and (v) aqueous separation chemistry. The EU CEG has developed co-operation between ISTC and FP5 and FP6 EU funded projects especially in the above area (ii), (iii) and (v) by organising joint meetings of BASTRA cluster with related ISTC projects and PARTITION cluster with related ISTC/STCU projects. In FP6, similar cooperation with EUROPART and EUROTRANS is being developed.

Cooperation between Euratom FP6 projects and US-DOE AFCI programme is also being fostered.

Conclusions

The research activities in the field of partitioning and transmutation under the EURATOM Fifth Framework Programme are nearly complete and have produced encouraging results. The research projects were regrouped into four clusters one on partitioning, and three on transmutation: basic studies, technological studies and fuel studies. These clusters and the design project formed a balanced programme on P&T that were co-ordinated by the ADOPT network. With a view to thoroughly integrate the EU research efforts, a European Research Area (ERA) initiative has been launched. The three projects selected (RED-IMPACT, EUROPART and EUROTRANS) in the area of P&T under FP6 are expected to contribute to making the ERA in P&T a reality. The collaboration between EU funded FP5 and FP6 projects and the ISTC/STCU projects on P&T is progressing satisfactorily.

Figure 1. **FP5 funded projects in the area of P&T under the umbrella of ADOPT network**



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TECHNICAL SESSION SUMMARIES

SESSION I

P&T Systems and Waste Management

P. Finck (ANL, USA) and A. Van Luik (DOE, USA)

Papers presented **6**

Effect of Advanced Fuel Cycles on Waste Management Policies
J-M. Cavedon (PSI, Switzerland)

Effective Application of Partitioning and Transmutation Technologies to Geologic Disposal
J. Ahn (LANL, USA) and T. Ikegami (Nuclear Cycle Development Institute, Japan)

Can Thermal Reactor Recycle Eliminate the Need for Multiple Repositories?
C.W. Forsberg et al. (ORNL, USA)

Repository Benefits of Partitioning and Transmutation
R.A. Wigeland and T.H. Bauer (ANL, USA)

Results on Transient Scenarios towards GEN IV Systems
F. Varaine, J-P Grouiller (CEA-Cadarache), M. Delpech,D. Warin (CEA-Saclay, France)

P&T Potential for Waste Minimisation in a Regional Context
M. Salvatores, J-P. Grouiller (CEA-Cadarache, France), M. Delpech (CEA-Saclay, France), E. Schneider (LANL, USA), A. Schwenk-Ferrero, H-W. Wiese, J.U. Knebel (FZK, Germany)

Session Themes (Presenters)

- Description of international collaborative work on P&T and its potential effects on waste disposal, including examples of early results on material flow-sheets and costs (J-M. Cavedon).
- Examples of evaluations of potential P&T effects on repositories in terms of toxicity and loading (J. Ahn).
- Examples of three potential paths forward:
  - Using thermal reactors until a fast reactor becomes available (C.W. Forsberg).
  - Steps in moving from current reactor fleet towards an all-GEN IV system (F. Varaine).
  - Regional cooperation in both waste minimisation and disposal (M. Salavatores).

Papers provided information regarding several aspects of potential P&T impacts

- Studies of the potential impacts of P&T on repositories focus on the potential difference in environmental impacts and on the dramatically enhanced efficiency of a repository that receives only fission products (removal and very long term storage of Cs and Sr was suggested to allow even greater more mass loading).
- Waste streams that are a part of P&T are being quantified and P&T costs are preliminarily being evaluated (much uncertainty).
- Evaluating benefits of P&T to repositories includes the delaying or eliminating the need for the need of further extension of the repository capacity or for an additional repository, in addition, supporting a societal decision to implement a P&T program requires evaluating costs and risks.

Path forward discussions provided long-term strategic insights

- Generation IV technologies are best suited to long term management of waste stockpiles.
- Nevertheless, if Generation IV technologies are delayed, a combination of current technologies (MOX) with a limited number of very advanced accelerator driven systems would also offer a solution to the waste issue.
- An alternative approach to classical P&T schemes exists, where all materials are recycled in LWRs.
- Provided spent fuel is cooled for (up to) several decades, quasi equilibrium of TRU inventories can be reached using such an approach.
- The LWR approach raises some practicality and potential non-proliferation questions that need to be addressed.

Chairmen's recommendations

- At the conclusion of the meeting, in the wrap-up session, this first session received much discussion because it was evidence of the P&T community's becoming fully integrated with its users at the back end of the fuel cycle.
- Discussants suggested it was a topic that should be inserted into future P&T Information Exchange Meetings. The Chairmen of Session I agree.
- The first paper in this session asked for help in completing an NEA document that takes a first, hopefully comprehensive, look at the integration of P&T fuel cycles and the waste management. The Chairmen of Session I recommend that organizations seriously look at the call for waste-management organization participants to provide safety evaluations in support of this study.

SESSION II

Partitioning Technology

J.H. Yoo (KAERI, Korea) and J. Uhlir (NRI, Czech Republic)

Papers presented **5**

Recent Progress in Actinide Partitioning at ITU
J-P. Glatz et al. (European Commission), T. Inoue et al. (CRIEPI, Japan)

Conversion of Oxide into Metal or Chloride for the Pyrometallurgical Partitioning Process
Y. Sakamura et al. (CRIEPI, Japan)

Study of Electrochemical Processes for Separation of the Actinides and Lanthanides in Molten Fluoride Media
R. Zvejskova et al. (NRI Rez, Czech Republic)

Processing of Spent TRISO-COATED GEN IV Reactor Fuels
B.B. Spencer et al. (ORNL, USA)

Partitioning of Cesium and Strontium from Dissolved Spent LWR Fuel using a Novel Crown Ether/Calixarene Solvent
J.D. Law et al. (INEEL, USA)

Overview

Five papers were presented during Session II on Partitioning Technology. Although no partitioning technology has been commercialized as yet, many intensive studies are being carried out in several countries. Recent progress in actinide partitioning at ITU, including an aqueous processing based on the DIAMEX-SANEX process and a pyrochemical process, was introduced.

The conversion of oxide into a metal or chloride for a pyrometallurgical partitioning was presented by CRIEPI. They effectively compared the process characteristics of such processes as a lithium-reduction, and electrochemical reduction, using lithium chloride or calcium chloride and the chlorination techniques by employing chlorine or zirconium chloride.

Electrochemical separation of actinide and lanthanide in a molten fluoride media was introduced by Czech-NRI, demonstrating the possibility of a uranium/thorium separation from fission products by an electrolytic deposition at the solid cathode via an electrorefining by employing a molten eutectic mixture of LiF-NaF-KF(FLINAK) as the electrolyte.

The experience of the crush-leach process for treating used TRISO-coated fuels was introduced by ORNL. Even although this process would be effective for removing the graphite block from fuel compacts, they pointed out the problems of the very fine particles of carbon in the filtration process and therefore it was stressed that the particle size should be carefully controlled during the milling.

The partitioning studies of cesium and strontium, aiming at reducing the heat load of spent nuclear fuels in a geological repository, by using the solvents of Crown Ether and Calixarene was presented by Nevada University. A synergistic extraction mixture for the simultaneous separation of cesium and strontium from dissolved spent nuclear fuel solutions was developed.

SESSION III

Fuels for Transmutation Devices

J. Wallenius (KTH, Sweden) and J-P. Glatz (ITU, Germany)

Papers presented **6**

Nitride Fuel and Pyrochemical Process Developments for Transmutation of Minor Actinides in JAERI
M. Akabori et al. (JAERI, Japan)

The Preliminary Performance Analysis of the Transmutation Fuel for Hyper
B-O. Lee et al. (KAERI, KOREA)

Characterisation of Actinide Alloys as Nuclear Transmutation Fuels
J.R. Kennedy et al. (ANL, USA)

Design Concepts and Process Analysis for Transmuter Fuel Manufacturing
G.F. Mauer (UNLV, USA)

Uranium Free Nitride Fuel Modelling, Fabrication, Characterisation and Irradiation
J. Wallenius (Royal Institute of Technology, Sweden)

Plutonium and Neptunium Conversion Using Modified Direct Denitration
L.K. Felker et al. (ORNL, USA)

Session summary

A number of presentations in this session have shown how important it is to consider all aspects relevant for the P&T scheme. The fuels will of course contain significant amounts of MAs and thus remote handling will be necessary, especially because P&T is based on a multirecycling scenario. It can be further assumed that an automation of the various processes involved contributes to reduce costs significantly.

The work carried out in the department of mechanical engineering at the UNLV on 3-D simulation helps to design processes under normal operation and incident (e.g. collisions at pin filling) conditions. The development applies for various types (composition and geometry) of fuels, e.g. dispersion fuels, ceramic fuels and metallic fuels in form of pellets or as dispersion fuel.

Another important issue is the link between separation (partitioning) and fuel fabrication and also in this case this conversion process will be of course necessary for any material recycling step.

The plutonium and neptunium conversion process developed at ORNL is using a modified direct denitration. In direct denitration a low-surface-area glassy product is obtained, therefore an additive is mixed with the uranyl nitrate solution to produce an oxide with desired ceramic properties comparable to oxide obtained by the ammonium diuranate (ADU) process. In fact it is important for the design of an efficient conversion process to obtain products, which are ready for fuel fabrication and at the same time to minimize any losses of actinides because this the prerequisite of an efficient P&T scenario (cf. also the article in the local newspaper, where it was claimed that P&T could make a deep underground waste disposal such as YUCCA MOUNTAIN redundant). It is planned to include in the near future all actinides.

The EC program CONFIRM dealing with the development of U free nitride fuels for application in ADS is a milestone in the P&T scenario development. Nitride fuels offer a promising potential in P&T scenarios and they were selected as a candidate fuel in the new EUROTRANS project. Carbo-thermic nitridation of PuO_2 and ZrO_2 powders is used for fabrication of (Pu,Zr)N; oxygen levels < 0.2 weigh % and pellet densities = 82% TD were achieved with this method. The thermal diffusivity and heat capacity of (Pu0.25,Zr0.75)N pellets were measured by CEA at Cadarache and pellets remained stable at T = 2 340 K under 1 bar of nitrogen.

The infiltration method developed at ITU is applied for the fabrication of (Am,Zr)N.

Am volatility during sintering is a key parameter to minimize losses and results show that a nitrogen atmosphere is much better in this respect if compared to an inert gas atmosphere. Problems related to the fuel reactivity need further investigation, but it could be shown here again, that nitrogen in the filling gas reduces the Am vapor pressure in the fuel pin.

Good quality fuels could be prepared and 4 (Pu, Zr)N pins were fabricated at PSI and send to Studsvik for irradiation in December 2003. In October 2004 Studsvik withdraw from the project and negotiations have started with NRG Petten to take over irradiation. If this transfer can be realized, a destructive PIE of irradiated CONFIRM pins will be included in the EUROTRANS project.

An other irradiation of sodium bonded (Pu,Am,Zr)N pins is scheduled to start in PHENIX in 2006 in the frame of the FUTURIX (DOE-CEA-ITU-JAERI collaboration) project. High temperature stability tests of (Pu,Zr)N will be made by the Bochvar Institute in Moscow and thermo-mechanical modelling of (Pu,Am,Zr)N irradiation in He, Na and Pb-Bi bonded pins, as well as for VIPAC are to be made by IPPE in Obninsk (MATINÉ project). All these activities underline the international interest in these new fuel materials.

SESSION IV

Transmutation – General

F. Varaine and M. Salvatores (CEA-Cadarache, France)

Papers presented **7**

Phenix: The Irradiation Programme for Transmutation Experiments
J. Guidez (CEA-Nuclear Energy Direction, France), D. Warin (CEA-Saclay), P. Chaucheprat (NUSYS, France), B. Fontaine et al. (PHENIX, France), A. Zaetta, F. Sudreau (CEA-Cadarache, France)

Pebble Bed Reactors for Once Through Nuclear Transmutation
P.T. León et al. (U.T.S – UPM, Spain)

An Assessment of Thermal-spectrum Transmutation Systems
C.G. Bathke et al. (LANL, USA)

Water and Lead-Bismuth Experiments: Fluent and Star-CD Simulation
A. Peña et al. (University of the Basque Country, Spain)

New Nuclear Data Libraries for Pb and Bi Isotopes
A.J. Koning et al. (NRG Petten, Netherlands)

Russian Programme of the Minor Actinide Nuclear Data Measurements and Evaluation
L.I. Ponomarev (Kurchatov Institute, Russia)

Status of Partitioning and Transmutation in India: Research, Development and Technology
B. Raj (Centre for Atomic Research, India)

Summary

The papers in this session did cover several different topics, without a defined focus:

- The first paper (presented by J. Guidez) summarized experiments to be performed in the PHENIX reactor in France. The reactor is presently shut down for standard refuelling operations, after ending the 51st cycle (which ran with an excellent load factor). Several experiment are or will be loaded, relevant to the demonstration of the waste transmutation:

  – In the physics (i.e. nuclear data assessment) area: the PROFIL – R and – M irradiations of pure Pu and MA isotope samples.

  – "Heterogeneous" once-through transmutation mode: ECRIX-B and-H experiments (Am on an inert matrix). CAMIX and COCHIX experiments (different fabrication processes).

– “Homogeneous” transmutation mode: METAPHIX (CRIEPI owner of the results), for full TRU recycling in a metal fuel. CAPRIX (high Pu content).

– Long Lived Fission Products: ANTICORP-1 (Tc-99).

The FUTURIX experiments (a CEA,DOE,ITU collaboration) will deal both with Pu-Am (nitride and oxide) fuels and with inert matrices of interest for Gen-IV GFR fuels.

- The second paper (presented by P. Leon) presented the concept of a pebble bed HTR in a once-through mode to reduce the radiotoxicity of MA produced in LWRs. The approach is to “break” the chain of successive neutron captures beyond Pu-242, in order to avoid the Am and Cm build-up. A parametric study was realized to optimise the ratio (capture Pu-242)/(fission Pu-239), by adjusting the kernel diameter of the TRISO particle. The simulation was performed with MCNP at BOL. A specific ratio was selected, but it was pointed out that future work should confirm the performance during irradiation, due to the expected significant variation of the spectrum and criticality management. The paper did also show some thermal-hydraulics simulations with the FLUENT code, to confirm the possible high outlet temperatures (for hydrogen production).

- The fourth paper (presented by A. Pena) gave an intercomparison of two CFD codes. The configurations chosen for that purpose, were related to, respectively, a water and a LBE experiment. The study was not intended to be a real analysis of the experiments, but rather a pure intercomparison of codes. With the parameters initially chosen for the analysis, some rather large discrepancies were found, and more work is needed for a full validation of models.

The fifth and sixth papers dealt with nuclear data issues:

- The paper by A. Koning *et al.* presented a complete evaluation of Pb and Bi isotopes, made with the model code TALYS. The results were compared to differential experiments, and the performance of the new evaluation was found to be very satisfactory, with an improvement with respect to the evaluations in the major data files. Some calculations of a few integral experiments were also presented, giving again rather satisfactory results. Work is planned to evaluate covariance matrices, but in the discussion a preliminary indication was given on (n,2n) and (n,n’) reactions uncertainty, which looks consistent with the requirement of 10-20% accuracy for design purposes.

- Finally, the paper by L. Ponomarev presented a proposal for a very comprehensive experimental program which should give access to most MA nuclear data in a wide energy range. Several Russian Institutions and installations would be involved, both for differential and integral experiments of high accuracy. This very valuable program should now find appropriate sources of financing, and there was the request to the major international laboratories involved in waste transmutation studies, to express their interest and support.

As a final comment, the remark was made that at this workshop, relatively few oral presentations were made in the crucial fields of materials and HLM technology, and it was recommended to take this point into account for future workshops.

SESSION V

Transmutation – ADS

P. D'Hondt (SCK-CEN, Belgium) and H. Oigawa (JAERI, Japan)

Papers presented **6**

EUROpean Research Programme

Transmutation Performance

R&D Activities in Japan

Reactor-Accelerator Coupling Experiment

Safety Issues of ADS

Spallation Target Systems

IP-EUROTRANS: A EUROpean Research Programme for the TRANSmutation of High-level Nuclear Waste in an Accelerator-driven System

J.U. Knebel and C. Fazio (FZK, Germany), H. Aït Abderrahi and P. D'Hondt (SCK-CEN, Belgium), G. Benamati and S. Monti (ENEA, Italy), E. Gonzalez (CIEMAT, Spain), S. Pillon (CEA-Cadarache, France) and D. Warin (CEA-Saclay, France)

Remarks:

- Gather design, coupling experiment, fuel, HLM technologies and nuclear data research.
- R&D programme focus on the need of the design.
- 23 M€ budget from EC.
- 5% of R&D budget is allocated to Education and Training.
- Start March of 2005 for a period of 4 years.

MA and LLFP Transmutation Performance Assessment in the MYRRHA Small-scale ADS
E. Malambu et al. (SCK-CEN, Belgium)

Remarks:

- Scoping calculations for a MYRRHA core containing 6 MA-assemblies close to the spallation target and assemblies containing LLFP in the reflector.
- Study shows the flexibility and ability for transmutation studies in a MYRRHA like installation.
- Measurable transmutation rates are for both MA and LLFP obtained after a irradiation cycle of one year.

R&D Activities on Accelerator-driven Transmutation System in JAERI
H. Oigawa et al. (JAERI, Japan)

Remarks:

- Reported on the design studies of a 800 MWth ADS.
- Related R&D foreseen in phase 2 (TEF) of J-PARC.
- Discussion was held on the scalability between the spallation target in the reference design and the one foreseen in TEF-T: through proton power density.

Overview of the AFCI Reactor-accelerator Coupling Experiments (RACE) Project
D. Beller (UNLV, USA)

Remarks:

- Only ADS related experiment going on in USA.
- Bridge between MUSE and TRADE.
- Low cost: 2-3 M.
- Now trough 2006.
- N-spectrum obtained is close to a spallation spectrum with a tail up to 30 MeV.

Safety Issues and Safety Indicators for Accelerator-driven Transmuters with Dedicated Oxide Fuels
W. Maschek et al. (FZK, Germany)

Remarks:

- Study performed for a 800 MWth ADT show that reactivity potentials can be higher than the subcritical margin.
- Safety indicators are proposed for ADT; applicability should be verified.
- Discussion:transients in ADT could be mitigated through the accelerator control.

Neutronic Analysis Studies of the Spallation Target Window for a Gas-cooled ADS concept
A. Abánades et al. (UPM, Spain), I. Gonçalves and P. Vaz (Instituto Tecnologico e Nuclear, Portugal)

Remarks:

- Work performed in framework of PDS-XADS.
- Neutron damage in the window is more important than the proton damage.
- Global effect of the neutron deposition from the core is negligible compared to the direct damage from the source.

Comments and recommendations

- Same kinds of technical challenges commonly exist in different system concepts.
  - Safety.
  - Material damage by protons and neutrons.
  - Thermal-hydraulics, corrosion, etc. for LBE.

Possibilities of international collaborations can be pointed out.

- In particular, basic idea for the safety evaluation of ADS should be discussed worldwide.
- Basic experiment for ADS by University is also under way in Japan. Such movement is welcome to breed next generations of scientists and engineers.

POSTER SESSION I

P&T Systems, Waste Management, Partitioning Technology and Fuels
Dominique Warin (CEA, France) and Joachim Knebel (FZK, Germany)

Papers presented	**16**
Thermodynamic data of actinides	1
Fuel behaviour	2
P&T programme	1
Effect on waste management	1
Waste immobilisation and deactivation	3
System studies	2
Separation process development	6

POSTER SESSION II

Transmutation
Enrique Gonzalez (CIEMAT, Spain) and Stefano Monti (ENEA, Italy)

Papers presented	**21**
Transmutation in thermal flux	3
Molten salt systems	2
Safety of ADS	2
Thermal-hydraulics of liquid metal cooled reactor	2
Material studies	2
Nuclear data measurements	3
Fuels and targets for ADS	2
ADS system studies	2
Detectors	2
Neutronic calculation	1

Annex 1

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