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Status and Trends of Nuclear Technologies

Report of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO)

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International Atomic Energy Agency

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FOREWORD

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was launched in the year 2000, based on a resolution by the IAEA General Conference (GC(44)/RES/21). INPRO intends to help to ensure that nuclear energy is available in the 21st century in a sustainable manner, and seeks to bring together all interested Member States, both technology holders and technology users, to consider, jointly, actions to achieve desired innovations. INPRO is taking care of the specific needs of developing countries.

This IAEA publication is part of Phase 1 of INPRO (Refs [1–3]). It intends to provide an overview on history, present situation and future perspectives of nuclear fuel cycle technologies. While this overview focuses on technical issues, nevertheless, the aspects of economics, environment, and safety and proliferation resistance are important background issues for this study. After a brief description about the INPRO project and an evaluation of existing and future reactor designs the publication covers nuclear fuel cycle issues in detail.

The publication was prepared from 2002 to 2004 by a group of experts from Canada, China, France, India, Japan, the Republic of Korea and the Russian Federation. The IAEA wishes to express appreciation to C. Ganguly, chairman of the experts group as well as to all authors for their presentations at the IAEA Technical Meeting on Innovative Nuclear Fuel Cycle Technologies (Vienna, Austria, April 2003,) and at the International Conference on Innovative Technologies for Nuclear Fuel Cycles and Nuclear Power (Vienna, Austria, 23–26 June 2003). Special thanks are expressed to H.G. Weidinger (expert in nuclear fuel technology, Siemens KWU, Nuremberg, Germany) for contributing to the organization, preparation, compilation and correction of the text of this report. The IAEA officers responsible for the organization of the activities of the experts group were Y. Busurin, a member of the International Coordinating Group (ICG) of INPRO and K. Fukuda of the IAEA Division of Nuclear Fuel Cycle and Waste Technology. The report was updated in 2008 to reflect the developments of nuclear energy since the creation of the original draft report in 2004.

It is expected that this documentation will provide IAEA Member States and their nuclear engineers and designers, as well as policy makers with useful information on status and trends of future nuclear fuel cycle technologies.

Due to the size of the full report it was decided to create a summary of the information and attach a CD-ROM in the back of this summary report with the full text of the report.

The IAEA officers responsible for this document are Y. Busurin of the Division of Nuclear Power, F. Depisch of the Division of Nuclear Power and C. Ganguly of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

CHAPTER 1
INTRODUCTION

1.1. About INPRO

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was established in 2001 in response to a resolution by the IAEA General Conference.

Objectives:

INPRO's objectives are:

- To help to ensure that nuclear energy is available to contribute, in a sustainable manner, to meeting the energy needs of the 21st century.
- To bring together technology holders and users so that they can consider jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles.

Missions:

INPRO's missions are:

- To provide a forum for discussion for experts and policy makers from industrialized and developing countries on all aspects of nuclear energy planning as well as on the development and deployment of innovative nuclear energy systems in the 21st century.
- To develop a methodology to assess innovative nuclear systems on a global, regional and national basis, and to establish it as an IAEA recommendation.
- To facilitate coordination and cooperation among Member States for planning of innovative nuclear system development and deployment.
- To pay particular attention to the needs of developing countries interested in innovative nuclear systems.

Recognition of INPRO

Since its establishment in 2001, INPRO has received recognition on various high level occasions, including the following: G8 Summit, Global Energy Security, St. Petersburg, 2006 and the US–Russia Strategic Framework Declaration by US president George W. Bush and Russian president Vladimir Putin, 2008.

History of INPRO

The 21st century will have the most competitive, globalized markets in human history, the most rapid pace of technological change ever, and the greatest expansion of energy use, particularly in developing countries. As IAEA Director General Mohamed ElBaradei said at the 50th IAEA General Conference, in September 2006, technological and institutional innovation is a key factor in ensuring the benefit from the use of nuclear energy for sustainability.

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was initiated in 2001, on the basis of a resolution by the IAEA General Conference in 2000

(GC(44)/RES/21). INPRO activities have since been continuously endorsed by resolutions by the IAEA General Conferences and by the General Assembly of the United Nations.

INPRO provides an open international forum for studying nuclear energy options, the associated requirements and the potential deployment of innovative nuclear energy systems in IAEA Member States. INPRO helps to make available knowledge that supports informed decision making during the development and deployment of innovative nuclear energy systems and assists Member States in the coordination of related collaborative projects.

INPRO’s initial activity (Phase 1, 2001–2006) focused on the development of an assessment method, called the INPRO methodology, to be used for screening an innovative nuclear system (INS), for comparing different INSs to find a preferred one consistent with the sustainable development of a given State and for identifying R&D needs. The INPRO methodology, tested for consistency and completeness, has been published and documented in three IAEA Technical Documents: Guidance for the Evaluation of Innovative Nuclear Reactors and Fuel Cycles (IAEA-TECDOC-1362), Methodology for the Assessment of Innovative Nuclear Reactors and Fuel Cycles (IAEA-TECDOC-1434) and Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems (IAEA-TECDOC-1575 Rev. 1), called the INPRO Manual, consisting of an overview volume plus a separate volume for each INPRO area of assessment.

INPRO Membership (2008)

As of December 2008, INPRO has 28 members: Argentina, Armenia, Belarus, Belgium, Brazil, Bulgaria, Canada, Chile, China, the Czech Republic, France, Germany, India, Indonesia, Japan, the Republic of Korea, Morocco, the Netherlands, Pakistan, the Russian Federation, Slovakia, South Africa, Spain, Switzerland, Turkey, Ukraine, the United States of America and the European Commission (EC).

FIG. 1.1. Members of INPRO in 2008.

1.2. About nuclear power

After the start on 8 December 1953, with the Atoms for Peace programme initiated by the USA, and a rapid growth in the 1960s and 1970s, mainly problems of acceptance, but also other concerns like safety of nuclear power plants (caused by accidents in TMI and Chernobyl), led to a slowdown of the growth of nuclear energy application worldwide towards the end of the 20th century. It is important to note that this slowdown is different in different areas of the world. It is most pronounced in many countries in Europe and in North America. There is considerable increase in the use of nuclear energy in the Far East and, at the moment, many developing countries are eager to step into a pronounced commercial use of nuclear energy if they could afford the investments for nuclear facilities and the corresponding infrastructure. Additionally, very recently in the western world there is a renewed interest in nuclear energy and sometimes the expression of a beginning ‘nuclear renaissance’ is being used within the worldwide nuclear community.

Assessment of history and the current situation of nuclear technology has to cover several aspects with regard to possible future developments, e.g. economics, proliferation resistance, protection of the environment, safety and sustainable development.

1.3. Outline of the report

The purpose of this report is to provide a general overview and to summarize knowledge accumulated in IAEA Member States in the area of advanced and innovative nuclear fuel cycle technologies. This report covers practically all different types of reactors and nuclear fuel cycle options with a special emphasis on innovative nuclear fuel cycle technologies, and summarizes technological approaches.

In Chapter 2 a short history, the current status, and future prospects of nuclear power plants (addressing all reactor types) are laid out.

Chapter 3 covers the same issues as Chapter 2, i.e. history, current status and prospects, however, focuses on technology of nuclear fuel cycle facilities. Additionally, the nuclear fuel cycle currently in use in selected countries is shortly presented.

Chapter 4 and Chapter 5 provide detailed information on the front end and on the back end of the fuel cycle, respectively.

Chapter 6 provides recommendations how to proceed in INPRO regarding nuclear fuel cycle issues thereby specifically addressing the needs of developing countries.

In the full report that is attached on a CD-ROM to this document there are two additional annexes not covered in this summary report: Annex A presents a Russian study on a sustainable global and national nuclear energy system; Annex B presents a summary of the status of multilateral nuclear fuel cycle centres.

CHAPTER 2
HISTORY, STATUS AND PROSPECTS OF NUCLEAR POWER

2.1. Short history of start of nuclear power

On 2 December 1942, within the US military project Manhattan, the first chain reaction occurred in the Chicago Pile-1 reactor under the leadership of E. Fermi. Up to today it is a big burden for any application of nuclear technology that its first use was the development of a nuclear weapon and its application in World War II.

After World War II, the US government encouraged the development of nuclear energy for peaceful civilian purposes. In 1953, president Eisenhower proposed his Atoms for Peace programme, which set the course for civil nuclear energy development in the western world.

In the USA, the first demonstration nuclear power plant (a pressurized water cooled reactor, PWR, 60 MW) built by Westinghouse started up in 1957. The first fully commercial plant (a boiling water cooled reactor, BWR, 200 MW) was built by General Electric (GE) and started up in 1960. In the middle of the 1960s, a kind of 'gold rush' of orders for nuclear power plants (PWRs and BWRs) occurred. Both reactor types used enriched ceramic UO_2 as fuel and light water as moderator and coolant. Additionally to light water cooled reactors (LWR) right from the beginning in the USA fast neutron reactors were developed and deployed[1]. The first nuclear reactor in the world to generate electricity (on a laboratory scale) was the sodium/potassium cooled fast neutron reactor EBR-1 in 1951.

In the UK, during the 1950s, a type of nuclear power reactor called Magnox was developed using metal natural uranium as fuel, gas as a coolant, and graphite as moderator. The first Magnox reactor started up in 1956, and in total 26 Magnox units were deployed. However, in the early 1960s, instead of Magnox an advanced gas cooled reactor[2] (AGR) was deployed using enriched ceramic UO_2 fuel. Finally, in the early 1990s, a nuclear power unit (PWR 1300 MW(e)) designed by Westinghouse was deployed at Sizewell in the UK.

In Canada, a power reactor (pressure tube heavy water reactor, CANDU) was developed and deployed in 1962 using natural uranium fuel and heavy water as moderator and coolant. The CANDU design continues to be refined up to today.

In France, a gas cooled nuclear reactor was developed similar to the Magnox design. A demonstration plant stated up in 1956 and commercial operation began in 1963. In the middle of the 1970s, France settled on standardized PWRs based originally on a licence agreement with Westinghouse.

In Germany, the first demonstration plant built in the 1950s was a reactor with natural uranium and heavy water as the moderator. The first commercial power plant was a BWR based on a licence with GE, which started up in 1961. Afterwards, BWRs as well as PWRs were deployed in Germany.

In Japan, a demonstration plant (BWR, 12.5 MW(e)) started up in 1963 based on a licence from General Electric. The first commercial nuclear plant starting up in 1966 was however a gas cooled reactor based on the UK Magnox design. Thereafter, commercial power plants deployed were either BWRs or PWRs (licence from Westinghouse).

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[1] A more detailed worldwide history of fast neutron reactors will be described in Section 2.5.

[2] A more detailed worldwide history of thermal neutron high temperature gas cooled reactors will be described in Section 2.4.

In the Russian Federation (the former Soviet Union), a 5 MW(e) graphite moderated and boiling water cooled reactor was commissioned in Obninsk in 1954. Thereafter, two types of nuclear power reactors started up in 1964, a BWR with graphite as moderator (100 MW, called RBMK) and a small PWR (210 MW, called WWER). Both types of reactors were developed further and deployed successfully thereafter.

In India, in 1969, two BWR units (160 MW(e)) were started in 1969 based on a turnkey contract with General Electric, USA. Next, a pressurized heavy water moderated reactor (90 MW(e), PHWR) was built in cooperation with AECL Canada in 1973 at Rajasthan (Rawatbhata). Since 1974, India has developed independently the PHWR design, which is the backbone of their nuclear power programme. Presently, 15 PHWR units (thirteen PHWR with 220 MW(e) and two PHWR with 490 MW(e)) are in operation.

The 1970s and 1980s were the two decades when the use of nuclear power for generation of electricity was rapidly spreading internationally to additional countries, such as Italy, Spain, Sweden, Romania, and Switzerland in Europe, to Argentina, Brazil and Mexico in Latin America, and to the Republic of Korea and China in the Far East. In this period, the commercial reactor sizes increased stepwise from 600, 900 up to 1300 MW(e).

2.2. Current status of nuclear power

Worldwide there were 439 nuclear units in operation as of end of 2007 [4]. These units produce approximately 14% of the world's electricity. The total number of operating nuclear units has been nearly constant since the beginning of the 1990s, i.e. although some of the older units were retired they were replaced by an equal amount of new plants going into operation. The worldwide nuclear capacity, however continued to increase up till today (although with a reduced growth rate after 1990), mainly due to the increased size of new plants but also due to the power up-rating of existing plants.

A majority (~88% of the total installed capacity of 371.6 GW(e) worldwide) of these reactors are light water cooled (264 PWR and 93 BWR units). About 6% of the worldwide installed capacity consists of heavy moderated reactors (42 HWR units, mostly CANDU type), and the rest is almost equally distributed (about 3% each) among gas cooled reactors (18 GCR, in the UK only) and light water cooled, graphite moderated water cooled reactors (16 RBMK, all but one in the Russian Federation). Two power stations with fast neutron reactors (one in France and one in the Russian Federation) are in operation as of end of 2007.

The largest number of operating nuclear units (103) and highest installed capacity (99257 GW(e)) is found currently (end of 2006) in the USA. Next is France with 59 units (and 63260 GW(e) installed), then Japan with 55 units (and 47587 GW(e) installed), followed by the Russian Federation with 31 units (and 21743 GW(e) installed). The Republic of Korea has 20 operating units (and 17454 GW(e) installed), the UK 19 units (and 10965 GW(e) installed), Canada 18 units (and 12810 GW(e) installed), Germany 17 units (and 20339 GW(e) installed), Ukraine 15 units (and 13107 GW(e) installed), and India 16 units (with 3577 GW(e) installed). Sweden and China have 10 operating units each (and 9097 and 7572 GW(e) installed, respectively), Spain has 8 (7450 GW(e) installed), Belgium 7 units (5824 GW(e) installed), the Czech Republic has 6 units (3523 GW(e) installed), Slovakia and Switzerland 5 units each (2034 and 3220 GW(e) installed, respectively), and Finland and Hungary have 4 units in operation (2698 and 1755 GW(e) installed, respectively). The remaining countries (10) with nuclear power have a maximum of 2 nuclear units operating. Geographically, the highest concentration of nuclear power plants is in Europe, followed by the eastern part of the USA.

The highest number of construction projects is currently (end of 2007) achieved in India with 7 units under construction (one unit is a fast breeder reactor). Next is the Russian Federation with 5 projects followed by China with 4 (in the Russian Federation and China one project is also an FBR), and the Republic of Korea and Japan with 3 projects each. Two projects were recently announced in Bulgaria and Ukraine, and one each is ongoing in Argentina, Finland, the Islamic Republic of Iran, the Republic of Korea, Pakistan and Romania.

Several countries have rather ambitious plans for extending the fleet of nuclear reactors within the near future. Official government planning of future additional nuclear power (as of end of 2007) within approximately the next 10 years shows about 16 new units in the Russian Federation, 14 units in China and in India, followed by Japan with 11 projected units. Other countries announced they intend replacing old units with new ones. A recent example is the UK that announced to include up to 10 new nuclear power plants into their energy plan.

Other nuclear countries announced end of 2007 to consider adding a limited number of nuclear units in the near future. Examples are: South Africa announced it will go out for bids to add several new nuclear units shortly. In one of the Canadian provinces a utility has announced plans to build a new nuclear unit, and also in the USA some nuclear utilities are expressing their interest in adding new nuclear plants within the next years. France announced it is going to start construction of a new nuclear unit (Generation III, EPR) in 2008, and Brazil is planning to start construction of one additional PWR at the ANGRA site. There are several developing countries that announced they are considering starting a nuclear power programme.

The current situation of nuclear power, as shown above, has been characterized as a kind of 'nuclear renaissance'. However, it has to be mentioned that there are several countries, exclusively located in western Europe, that keep up their nuclear phase out policy defined in the 1990s despite this 'nuclear renaissance' going on presently.

In the following sections a short overview will be provided on development of water cooled reactors, gas cooled reactors and fast neutron reactors (Refs [5–8]).

2.3. Development of water cooled reactors

History of the development and deployment of water cooled reactors (WCR) till the 1980s has already been outlined in Section 2.1. In the following sections, the current situation of WCR development (taking into account the last decades) will be laid out.

Three types of WCR were developed since the 1980s worldwide:

- large size (1000 to 1750 MW(e)) light water moderated and cooled power reactors;
- small to medium size (25 to 300 MW(e)) light water moderated and cooled reactors most of them with an option for non electrical applications (cogeneration), e.g. process heat for desalination or district heating;
- medium size (300 to 1200 MW(e)) heavy water moderated and heavy or light water cooled power reactors.

Large size light water cooled reactors

In China, Europe, Japan, the Republic of Korea, and the USA several large size light water cooled and moderated reactors were developed and some of them are under construction or went into operation already. The following table provides an overview of these water cooled reactor designs.

Table 2.1. DEVELOPMENT OF LARGE SIZED WATER COOLED REACTORS

Origin	Type	Name	Power (MW(e))	Status (2007)
China	PWR	CNP	1000	Designed
France/Germany	PWR	EPR	1750	Under construction
Germany	BWR	SWR	1200	Designed
Japan/USA	PWR	APWR	1750	Designed
Japan/USA	BWR	ABWR	1380	Operating
Japan	BWR	ABWR II	1700	Designed
Republic of Korea	PWR	KNSP/OPR	1000	Operating
Republic of Korea	PWR	APWR	1400	Under construction
Russian Federation	PWR	WWER V392	1000	Under construction
Russian Federation	PWR	WWER V448	1500	Designed
Russian Federation	PWR	AES-2006	1200	Under construction
Sweden/USA	BWR	ABWR90+	1500	Designed
USA	PWR	System 80+	1000	Designed
USA	PWR	AP1000	1000	Designed

Small and medium sized light water cooled reactors

In Argentina, China, Japan, the Republic of Korea, the Russian Federation and the USA a large number of designs of small and medium water cooled reactors is under development. Most of these designs are intended to be used for cogeneration and a few exclusively for non-electrical applications, such as process heat for desalination or district heating. Some of these designs have reactors that do not need on-site refuelling.

Only one of these small water cooled reactors, the Russian KLT-40 (a floating twin PWR reactor system with 2 × 135 MW(th)) is currently under construction. Examples of completed designs of a medium sized integral PWR[3] are the SMART reactor (330 MW(e)) designed in the Republic of Korea, the CAREM reactor (25 to 330 MW(e)) designed in Argentina, the Russian design VBER-300 (330 MW(e)), the multinational designed IRIS (330 MW(e)) and the Chinese design of a heating reactor called NHR-200. Studies on similar reactor designs are also performed in China, Japan and the USA.

Heavy water moderated reactors

Heavy water moderated reactors are primarily developed in Canada and in India.

Canada is continuously developing its CANDU design, CANDU-9 (900 MW(e)) being the latest design with a heavy water moderated and cooled reactor. Already, the next generation

[3] One of the important features of an integral PWR is the location of steam generators inside the reactor pressure vessel.

of CANDU reactor has been designed and is called Advanced CANDU Reactor (ACR). The ACR has a capacity of 700 MW(e) and uses heavy water as moderator, but light water as coolant. A version of the ACR under development will have a capacity of 1200 MW(e). A long term development project is the design of a CANDU with supercritical water conditions in the coolant.

India has developed the design of an advanced type of heavy water moderated and light water cooled reactor called AHWR (300 MW(e)) suitable for the use of thorium as fertile material.

2.4. Development of high temperature gas cooled reactors

First, a short history of the development of the high temperature gas cooled reactor will be presented and then, some ongoing national development programmes will be laid out.

2.4.1. Short history of high temperature gas cooled reactor development

The development of high temperature gas cooled reactor (HTGR) designs began in the 1950s shortly after the start up of WCR. An important role in the development of HTGR was played by the 20 MW(th) Dragon research reactor in the UK. It operated from 1964 through to 1977 within a framework of the OECD/NEA international collaboration as a productive research tool for the development of HTGRs.

Following the Dragon reactor, in the 1960s the demonstration reactors AVR (15 MW(e)) and the Peach Bottom HTGR (40 MW(e)) were constructed and successfully operated in Germany and USA, respectively. Both reactors used graphite as a moderator and helium as coolant in the core.

Commercialization was approached via the FSV-HTGR (330 MW(e)) that operated in the USA from 1976 to 1989 and by the THTR-300 (300 MW(e)) in Germany that operated from 1971 to 1989. In both reactors thorium was used as a fertile material in their core design. Especially in Germany this type of reactor was intended to be used for cogeneration, i.e. producing process heat for hydrogen production or other industrial applications such as fossil fuel upgrading.

No new power reactors of the type HTGR were build after 1976, but a lot of concepts were developed in Europe and the USA till the end of the 1980s. Concepts include the German HTR-500, the Russian VG-400 and the US HTGR-SC. The modular concept was followed by direct cycle gas turbine modular design which could reach thermal efficiencies as high as 48%.

2.4.2. Ongoing national high temperature gas cooled reactor development programmes

Currently, active development activities in regard to the high temperature gas cooled reactor (HTGR) are ongoing in several countries, primarily in China, France, Japan, the Republic of Korea, South Africa, the Russian Federation and the USA.

China is operating a pebble bed modular high temperature gas cooled reactor (PBMR) with 10 MW(th) since 2003. This test reactor is called HTR-10. It uses coated particles in graphite balls similar as the German AVR, and has a steam cycle for energy conversion. R&D of a direct helium cycle system is ongoing with the goal to design a commercial version of the HTR-10 with 160 to 200 MW(e) similar to the design pursued in South Africa.

In France, significant R&D and design studies are performed to develop a very high temperature gas cooled reactor (VHTR) to be used for cogeneration (electricity plus heat).

In Japan, since 1998, a high temperature test reactor (HTTR) with 30 MW(th) is in operation. It reached a helium outlet temperature of 950°C in 2004. A facility for hydrogen production is tested separately and is planned to be thermally (10 MW) coupled to this reactor. R&D is ongoing to develop a commercial version of the reactor with 600 MW(th) and a direct energy conversion cycle.

The Republic of Korea is performing intensive R&D and conceptual studies of a very high temperature reactor (VHTR) within the Generation IV initiative (to be described in Section 2.5.2).

South Africa is performing R&D on a pebble bed modular gas cooled reactor (PBMR) since 1993. The design has been finalized in the meantime and key components have been ordered. Construction is intended to start in 2009. A helium test facility (HTF) started up in 2004 and should support the final design of the direct cycle gas turbine.

The Russian Federation and the USA have teamed their development efforts for a modular helium cooled reactor (MHR) since the 1990s. The design of a MHR has been completed in 2001 and construction of a prototype is scheduled to start in the Russian Federation in 2010. Initially, it is intended to burn weapons plutonium but later it should be used for burning all actinides and even for hydrogen production. Licensing of this reactor in the USA is planned.

2.5. Development of fast neutron reactors

As has been done for the high temperature gas cooled reactor, first a short history of fast neutron reactor development will be presented and then some ongoing national programmes will be laid out.

2.5.1. Short history of fast reactor development

During the first 20 years of their existence, the systems of fast neutron reactors and water cooled thermal neutron reactors advanced side by side. The first fast reactor was called Clementine and started up at Los Alamos (USA) in 1946 with a power of 150 kW and mercury as a coolant. As already mentioned before, the first nuclear reactor in the world to generate electricity (on a laboratory scale) was also a fast reactor: the EBR-1 in the United States of America in 1951. A fast reactor called BR-2 (100 kW) also cooled with mercury started up in 1956 in Obninsk (the former Soviet Union) supplying electricity to the research centre.

Around the 1960s, four experimental fast reactors (DFR in the UK, RAPSODIE in France, EBR-2 in the USA, and BOR-60 in the Russian Federation) of about the same power went critical. After the DFR, that used a mixture of sodium/potassium, all following fast reactors used sodium as a coolant. The oldest and largest of these four, the DFR (72 MW(th)), was successfully operated over 18 years; a similar operation time was achieved by the RAPSODIE (40 MW(th)) later on. The EBR-2 (62 MW(th)) was in operation for 30 years and the BOR-60 (55 MW(th)) is even still in operation now.

The first prototype fast reactor for power production was the EFFBR (200 MW(th)) that operated from 1963 to 1966 in the USA. It was shut down after a fuel melting incident in 1972.

From 1972 to 1974, three prototypes of comparable size were successively brought into operation: BN-350 (750 MW(th)) in the former USSR (now in Kazakhstan), PHENIX (560 MW(th)) in France and PFR (650 MW(th)) in the UK. The second one, PHENIX, is still in operation in France. BN-350 and PFR were finally shut down in 1999 and 1994, respectively. A 400 MW(th) fast flux test facility (FFTF) operated in the USA from 1982 to 1992.

In Germany, an experimental sodium cooled fast reactor called KNK II (60 MW(th)) operated from 1977 to 1991. A commercial version of this fast reactor called SNR-300 was constructed but never started up due to political decisions.

In Japan, in 1977, the experimental fast reactor called JOYO (140 MW(th)) started up and is still in operation today. It was followed by a demonstration fast reactor called MONJU (714 MW(th)) in 1994. Because of a sodium leak incident in 1995 this reactor was shut down and after significant design upgrading is supposed to start up again in 2008.

In India, in 1985, an experimental fast reactor called FBTR (40 MW(th)) started up and is still in operation today.

Large scale demonstration plants with fast reactor were build in the Russian Federation (1980), the so-called BN-600 (600 MW(e)) and in France (1986) the SUPERPHENIX (1240 MW(e)). This French fast reactor was shut down in 1994, but the Russian BN-600 continues to operate successfully up to today.

The motivation for building fast reactors has progressively changed. At the outset, the main objective for developing the fast reactor was breeding in order to conserve uranium resources. In reality, however, uranium remained abundant and cheap, mainly because the growth rate of nuclear energy was lower than had been expected.

Consequently, the use of fast reactors in a ‘burner’ mode for managing excess plutonium gained in importance and remains today a particular focus of fast reactor R&D activities. Moreover, the desire to further optimize the back end of the fuel cycle including the disposal of high level waste has recently been stimulating an increasing interest in extending the application of the fast reactor from the burning of plutonium to the burning (transmutation) of all transuranic actinides, known as minor actinides (MA: Np, Am, Cm).

2.5.2. Ongoing national fast reactor development programmes

As outlined in the section above, in four countries (the Russian Federation, France, Japan, and India) experimental and/or demonstration plants with a fast reactor are in operation today. Additionally, in 3 countries (China, India and the Russian Federation) construction of a fast reactor is currently ongoing.

China has an ambitious programme for development of FR. The experimental fast reactor with sodium cooling called CEFR-25 (25 MW(th)) is under construction and could reach first criticality in 2009. Design and analytical studies are performed for a prototype fast reactor called CPFR (600 MW(e)) to be deployed in 2020, to be followed by a commercial fast reactor called CCFR (1000 MW(e)) around 2040.

France has a broad national R&D programme covering almost all advanced reactor designs and corresponding fuel cycles in discussion worldwide. Regarding fast reactors France is pursuing two options: a fast reactor with sodium and alternatively with gas as the primary coolant. Energy conversion systems studied include intermediate and direct gas cycles for both reactor types. An important role in their fast reactor development programme is the sodium cooled PHENIX reactor which they use for experiments with an emphasis on nuclear fuel development for transmutation of transuranics. Analytical and design studies for a large scale (1500 MW(e)) commercial fast reactor with sodium (SFR) and gas (GFR) cooling are and have been performed. An important milestone of the SFR programme is the planned construction of a prototype SFR with 250 to 600 MW(e) and of the GFR programme the erection of an experimental technology demonstration reactor (ETDR); both plants are scheduled to be commissioned in 2020.

India has a unique experimental fast neutron reactor FBTR (40 MW(th)) in operation since the 1980s that uses uranium/plutonium carbide as fuel (all other fast reactors in operation use U/Pu oxide as a fuel). In addition, a prototype fast breeder reactor PFBR (500 MW(e), sodium cooled, pool type) is under construction to be commissioned in 2010.

Japan has one experimental fast reactor called JOYO in operation and is expecting restart of its demonstration fast reactor called MONJU shortly. Japan's R&D programme regarding fast reactors is called FaCT and covers all possible types of primary cooling for fast reactor: Na, He, Pb + Bi and H_2O. All sizes of reactors, e.g. large, medium and small are under investigation. The development programme was recently focused on an advanced fast reactor with sodium cooling. Around 2025 a demonstration and 2040 commissioning of a commercial fast reactor is foreseen.

The Republic of Korea's development programme on fast reactors covers Na and Pb-Bi as coolant. The design concept of a sodium cooled fast reactor called KALIMER is being developed with a power output of 150 and 600 MW(e). The corresponding R&D project for a Pb-Bi cooled fast reactor is called PEACER.

The Russian Federation has two operating fast reactor, the experimental fast reactor BOR-60 and the demonstration fast reactor BN-600; additionally the prototype fast reactor BN-800 is under construction. Design studies are performed for a large commercial fast reactor called BN-1800. Based on long term experience with Pb-Bi cooled naval reactors the development of commercial power fast reactor using this coolant called BREST (300 and 1200 MW(e)) and SVBR-75/100 is being pursued. Conceptual design studies of a supercritical water cooled fast reactor are underway.

In the USA, a broad development programme for fast reactor technology has been initiated recently, partly embedded within international initiatives for R&D collaboration, such as the Generation IV International Forum (GIF) and Global Nuclear Partnership (GNEP). Around 2040, in the USA the commissioning of a commercial fast reactor is envisaged.

2.6. Multinational programmes for development of advanced reactors

Generation IV International Forum (GIF) [9] was founded in 2001 by the US Department of Energy together with Argentina, Brazil, Canada, France, Japan, the Republic of Korea, and the UK. In the meantime, South Africa, Switzerland, the European Union, the Russian Federation and China joined. An international team of experts selected 6 types of advanced nuclear reactors (called Generation IV reactors) and their fuel cycle technology to be developed jointly within the next 15 to 25 years: A gas cooled fast reactor (GFR), a sodium cooled fast reactor (SFR), a lead cooled fast reactor (LFR), a molten salt reactor (MSR), a supercritical water cooled reactor (SCWR), and a very high temperature reactor (VHTR). Most of the systems are envisaged to employ a closed fuel cycle to maximize the efficiency of use of resources and minimization of high level waste. The national development programmes of the countries participating in GIF are synchronized to this international activity.

The Global Nuclear Energy Partnership programme (GNEP [10]) was launched in 2006 by the US Department of Energy together with China, France, Japan and the Russian Federation; in 2007, Australia, Bulgaria, Ghana, Hungary, Jordan, Kazakhstan, Lithuania, Poland, Romania, Slovenia and Ukraine joined. The goals of GNEP include the expansion of nuclear power in the USA and worldwide, an aggressive plan to manage spent fuel in the USA, the development of advanced nuclear fuel cycles including advanced reprocessing technologies with enhanced proliferation resistance and less waste, a fuel service programme enabling nations to acquire nuclear energy while limiting proliferation risks, the development of new types of reactors such as small reactors for export and advanced burner (fast neutron) reactors (ABR) to reduce transuranics (actinides) in nuclear waste, and the reduction of separated civilian plutonium. GNEP will move the USA from a once through fuel cycle to a closed or recycling fuel cycle. The development of the ABR is the type of fast reactor to be developed within the USA.

The European Commission funds studies and experiments on the future of nuclear systems in accordance with their policy on security of energy supply. Those studies focus on energy production and waste management including partitioning and transmutation. The European Commission has joined GIF in 2006 investigating gas cooled fast reactors, lead cooled reactors and very high temperature reactors.

Other examples of bilateral cooperation are the USA/Russian Federation, the France/Japan and the USA/Japan collaborative programmes covering all types of reactor concepts defined in GIF.

2.7. Development of very advanced reactor systems

The concept of two very advanced reactor systems is currently evaluated again (the concept was investigated already as early as the 1940s, but dropped at that time) worldwide because they offer substantial progress with regard to waste minimization and proliferation resistance. The two systems are the molten salt reactor (MSR) and the accelerator driven systems (ADS).

ADS will be discussed further in Section 5.3 together with the fuel cycle technology of partitioning and transmutation (P&T) of transuranics and fission products.

The first MSR was the aircraft reactor experiment (ARE, 2.5 MW(th)) designed to be used as an engine for a military aircraft in 1954 at the Oak Ridge National Laboratory (ORNL) in the USA. It was continued by the molten salt reactor experiment (MSRE) in ORNL with a 7.4 MW(th) test reactor simulating an epithermal thorium breeder reactor.

Recently, conceptual design studies of the MSR are being performed again in several countries, e.g. the Russian Federation, France and the USA, mostly within the GIF international project.

2.8. Perspectives of nuclear power reactors

Today, studies on the regional and worldwide growth of the energy demand and the role of nuclear energy within this growth always provide a set (or family) of alternative scenarios.

Within INPRO several SRES[4] scenarios have been evaluated [1] for the future worldwide growth of energy demand and supply by all possible energy sources including nuclear. Although there are significant differences shown for different geographic areas in the world, all SRES storylines evaluated project a significant total increase of nuclear power worldwide within the 21st century, i.e. nuclear energy will be needed inevitably to satisfy the future global demand for energy. The evaluation [1] also showed that during the 21st century the share of nuclear power could not only be kept constant but even increased dramatically if its

[4] SRES stands for Special Report on Emission Scenarios commissioned by the Intergovernmental Panel on Climate Change.

economic competitiveness would be further improved continuously (concept of learning rates) in comparison with alternative energy sources.

As already indicated in Section 2.2 there are increasing indications that the present global stagnation of nuclear power may be overcome due to various reasons such as the uncertainties of the international supply of oil and gas, and growing concern of impact of green house gases on the climate.

IAEA projections up to the year 2030 show a possible increase of installed nuclear power from 370 GW(e) currently (2006) to 420 GW(e) for the low case and up to more than 600 GW(e) in the high case.

For the next decades most probably the water cooled reactors with further enhanced safety and increased competitiveness will be the backbone of nuclear power generation, i.e. electricity production.

CHAPTER 3
NUCLEAR FUEL CYCLE OPTIONS

3.1. Short history of nuclear fuel cycle strategies

Already in the late 1940s, the complete nuclear fuel cycle with uranium/plutonium was established in several countries consisting of the following steps: mining, milling, conversion, enrichment, fuel fabrication, energy conversion (reactor), reprocessing, and waste storage (but no final depository of waste). However, this nuclear fuel cycle was created and applied within military (nuclear weapons) programmes and only later on (1950s and 1960s) transformed into civilian (commercial) application of nuclear power.

In the 1970s and 1980s, nuclear power grew rapidly in the USA, Japan and in Europe (e.g. France, Italy, Germany, Spain, Sweden and the UK). In these countries the nuclear fuel cycle strategy changed considerably during that period of time. Due to the oil embargo in the beginning of the 1970s security of energy (electricity and fuel) supply reached highest priority to be achieved by introduction of fast breeder (FBR) and high temperature gas cooled reactors (HTGR) and fuel recycling. However, due to declining growth rate of nuclear power, increased efficiency of thermal water cooled reactors (via higher burnups and availability), decreasing uranium costs, and assurance of oil supply, the need for HTGR and fast reactor was no longer seen towards the end of the 1980s. The thermal reactor with a once through and MOX mono-recycling became the dominant nuclear fuel cycle.

In some countries the strategy of the commercial nuclear fuel cycle was also changed due to political reasons. For example, in the USA, based on a government decision in the 1970s, commercial reprocessing and recycling of plutonium (in mixed oxide fuel, MOX) was stopped, including the corresponding programmes for commercial fast reactor deployment. However, basic R&D programmes continued in the USA focusing on development of advanced fuel designs and conceptual designs. During the same time primarily in Europe, MOX technology was developed to a mature technology and commercially applied in several light water power reactors.

3.2. Current status of nuclear fuel cycle technology

Presently, uranium is the basis for commercial nuclear power production. An attractive additional resource of fertile material will be the use of thorium.

There are, in principle, two kinds of nuclear fuel cycles: a closed and an open nuclear fuel cycle. In the first one, all fissile and fertile material in spent nuclear fuel (SNF) is recycled, i.e. returned to the reactor for further energy conversion; waste to be deposited contains only the fission products and minor actinides, i.e. materials that cannot be used within the technology. In the second one, the fuel is used only once in the reactor and the SNF is deposited finally regardless how much fissile and fertile material is left in the SNF. In reality, there are several mixtures between these two principal kinds of a nuclear fuel cycle used in countries, like partial recycling of SNF, recycling of all fissile material, and recycling of all actinides and even fission products.

Based on an IAEA survey in 2001 [11] the following countries used a (mostly partially) closed nuclear fuel cycle: Belgium, China, France, Germany, India, Japan, the Netherlands, the Russian Federation, Switzerland, and the UK. Since 2001, several countries have stopped reprocessing/recycling of SNF, i.e. Belgium, Germany and the Netherlands have moved to an open fuel cycle. An open fuel cycle was used in 2001 in Canada, Finland, Sweden and the USA. In the meantime, the USA has announced (GNEP) that it will be considering a closed fuel cycle.

3.3. Trends of nuclear fuel cycle technologies

A global study performed by the Nuclear Energy Agency in 2001 [12] on trends of nuclear fuel cycle development came up with the following results:

- Short term development will concentrate on cost reduction to increase the competitiveness.
- Medium term development will focus on solutions of the back end of the nuclear fuel cycle, primarily the treatment of long living transuranics in SNF.
- Fuels suitable for reactors with non electrical applications will be developed in parallel to the reactor development.
- Only long term (> 20 years) development will enable introduction of partitioning and transmutation technologies, fully closed nuclear fuel cycle with fast reactors including molten salt reactors, and thorium on a commercial basis.

Additional national and international studies on availability of fissile material for global nuclear energy systems have concluded that towards the end of this century (assuming constant nuclear power production) the current nuclear fuel cycle using thermal neutron reactors and U (and a limited amount of Pu) as fuel will exhaust existing resources, i.e. the introduction of fast breeder reactors and a closed nuclear fuel cycle seems inevitable to achieve a sustainable solution.

To extend the capability of existing resources of uranium (and delay the introduction of commercial fast reactors) the fuel efficiency of thermal reactors could be improved by:

- higher burnups;
- decreasing the lower tails of the enrichment process;
- increasing thermal efficiency of reactors;
- extending the recycling of Pu;
- start using Th as fertile material.

3.4. Advanced nuclear fuel cycle technologies with a potential industrial deployment in ~25 years

Currently, there are several concepts of advanced nuclear fuel cycles developed that could be implemented within the next 25 years. A few examples are given below.

DUPIC [13]

One advanced nuclear fuel cycle technology to be implemented within the next 25 years, intensively studied in the Republic of Korea together with Canada and the USA, is the Direct Use of Spent PWR fuel in CANDU reactors (DUPIC). It converts SNF from PWRs via a thermal mechanical process into CANDU fresh fuel.

Its main advantages are increased proliferation resistance, and a reduction of waste from CANDU reactors. Its main disadvantages are the need of remote handling of CANDU fuel fabrication and of fresh CANDU fuel, and the limited savings of cost and resources.

Thorium fuel cycles [14]

Globally, there are 1.2 million tons of known resources of thorium, about 90% in the following countries: Australia (25%), India (24%), Norway (14%), the USA (13%), Canada (8%), South Africa (3%) and Brazil (1%).

In the past, Th has been used in several demonstration reactor projects, e.g. in LWR (USA), HTGR (USA, Germany, UK), and in MSR (USA). However, in India Th assemblies have been inserted in several commercial PHWR to flatten the neutron flux in the initial core during start up.

Regarding Th utilization, thorium-based coated particle fuel technology seems to have the highest potential in HTGR. For reprocessing the process THOREX, developed in ORNL, USA, appears technically feasible.

The main physical advantage of using Th as fertile material in a nuclear fuel cycle is its higher neutron yield per neutron absorbed, which enables higher conversion or breeding rates (producing ^{233}U from Th) compared to ^{238}U (producing Pu). One technical disadvantage is the necessity of sufficient shielding during recycling and fuel fabrication using ^{233}U due to radiation levels (high gamma and beta, caused by ^{232}U decay products), and the possibility of chemically separating the fissile element ^{233}U from Th in spent nuclear fuel poses some proliferation risk (comparable risk to a U/Pu nuclear fuel cycle).

In addition to India that assigned highest priority to commercial use of Th in their national nuclear fuel cycle (to be used in PHWR, fast reactor and AHWR in the near future), several other countries are currently performing R&D of Th fuel in short and/or long term programmes.

Reduced moderation LWR (RMWR) MOX Nuclear Fuel Cycle

LWRs that use a tight fuel rod lattice in the core resulting in reduced moderation and consequently in a hard (fast) neutron spectrum could achieve high U/Pu (or Th/U) conversion rates (even > 1).

Primarily in Japan (but also in the Russian Federation and the USA), conceptual design studies (of such LWR reduced moderation cores) are being performed showing promising results, and indicating a possible implementation with the next 25 years.

Advanced fuel cycle initiative (AFCI)

In 2003, in the USA, a broad R&D programme was established to accomplish a transition of the current open fuel cycle to the advanced nuclear fuel cycle to be used in advanced types of reactors as defined in the Generation IV initiative. The advanced fuel cycles will reduce high level waste volume, greatly reduce long lived and highly radiotoxic elements, and reclaim valuable energy content in SNF. The main programme includes system analysis and development of technologies for separation and transmutation.

Additional examples of advanced nuclear fuel cycles

Within the Generation IV initiative nuclear energy systems consisting of LWR and HTGR (pebble bed or prismatic uranium oxide fuel) with a once through cycle are studied in the USA as a near term option. Other long term concepts studied consist of a combination of LWR and fast burner reactors to be converted to fast breeder reactors later.

In the Russian Federation together with experts from Japan and the USA the concept of the BREST reactor and its advanced nuclear fuel cycle has been studied. It uses a closed nuclear fuel cycle based on U/Pu nitride fuel. Fuel reprocessing and fabrication facilities should be co-located near this type of reactor.

CHAPTER 4
FRONT END OF NUCLEAR FUEL CYCLE

In this chapter history, status and perspectives of all issues (e.g. exploration) and facilities (mining, milling, conversion, enrichment and fuel fabrication) of the front end of the nuclear fuel cycle of all reactor types (e.g. thermal and fast neutron reactors) are shortly laid out [15].

4.1. Uranium resources

Since a peak in 1997, a continuous decrease of uranium exploration efforts occurred worldwide till about 2003. After that year a sharp increase was observed of the money spent on exploration domestically and non-domestically. This clearly demonstrates that the uranium mining industry was reacting to the actual and projected increase in demand and consequently in price of uranium. In 2007, most money was spent by Canada, followed by the Russian Federation, France and the USA.

Commonly, there are three categories of 'conventional' uranium resources defined according to how well they have been proven: identified, prognostic and speculative resources. Additionally, in all three categories the amount of available uranium is distinguished according to its estimated production cost, again in three categories: < 130, < 80 and < 40 US$/kg. In 2005, the identified resources of uranium with a production cost < 130 US$/kg were 4.7 million tons (3.8 million tons for < 80 US$/kg, and 2.7 million tons for < 40 US$/kg). The prognosticated and speculative resources were estimated to be 2.5 and 4.6 million tons, respectively.

As of 2007, identified resources of uranium are mostly located in Australia (~23% of worldwide resources), Kazakhstan (~15%), the Russian Federation (~10%), South Africa (~8%), Canada (~8%), the USA (~6%), Uzbekistan (~6%), Brazil (~5%), Namibia (~5%), and Niger (~5%).

There are also so-called non-conventional uranium resources which have not been used on a commercial scale up till now: 4 billion tons in sea water with recovery costs of about 300 US$/kg, and 22 million tons in phosphates.

It is to be mentioned that in addition to natural resources there is a considerable amount of fissile/fertile material available from processed uranium and spent nuclear fuel (SNF) in nuclear facilities. This material includes military high enriched uranium and plutonium, enrichment tails, and reprocessed uranium (RepU) and could provide the complete fuel for the existing fleet of reactors (439) for a period of about 10 years. Last but not least, there is a large amount of stored SNF that, if recycled, could also be used as a source to provide fuel to the existing reactors for about 6 years.

4.2. Mining and milling of uranium

There are three basic processes of mining used today: Underground mining, open pit mining and in situ leaching. The first two methods are used in about 60% of all uranium production sites worldwide in 2007.

Uranium recovery from phosphates is a mature technology but not used commercially today because of its high costs from 60 to 100 US$/kg. To recover uranium from seawater, only a pilot scale plant has been shortly operated by Japan using an adsorption process.

At present, the main uranium ore producing countries (world production in 2007 was ~43000 t·U) are Australia (25% of world production), Canada (19%), Kazakhstan (13%), Niger (9%), Namibia (8%), the Russian Federation (8%), Uzbekistan (6%), and the USA (5%). In addition uranium mining and milling is carried out on a commercial scale in Brazil (0.8%), China (1.8%), the Czech Republic (1.0%), Germany (0.2%), India (0.6%), Pakistan (0.1%), South Africa (1.8%), Ukraine (2.0%), and the USA (2.2%).

Historically, the world production of uranium declined from 1988 (about 60000 t·U) till about the middle of the 1990s by about 50% (to about 30000 t·U) and remained almost constant thereafter till about 2003, when it started to increase again and reached a level of about 43000 t·U in 2007.

However, world demand for uranium increased almost linearly from 1988 (about 50000 t·U) to about 70000 t·U in 2007. The gap between production and demand was closed by so-called secondary sources (e.g. stock piles of enriched uranium, reprocessing of spent fuel, re-enrichment of enrichment tails, etc.) as discussed in the previous section.

Looking into the future till about 2030, although demand for uranium still is projected to grow steadily, the projected natural uranium production (committed, planned and prospective) capacity is to increase rapidly and satisfy the complete uranium demand around 2009, and may even surpass demand.

It is interesting to note that as late as 2004 the projections of demand and supply showed an ever increasing gap between production and demand that could have created a real shortage of uranium around 2010. But as discussed already in Section 4.1 in parallel to exploration also uranium production increased appropriately to cover this gap.

4.3. Conversion of uranium

Within a nuclear fuel cycle the term conversion means purifying natural uranium concentrate in the form of U_3U_8 to get U_{metal}, UO_2, or UF_6. Natural U_{metal} is used for fuel fabrication of Magnox reactors in the UK, Natural UO_2 for fabrication of PHWR fuel, and UF_6 is needed for the enrichment process of uranium for LWR fuel.

The worldwide capacity for conversion is about 130000 t·U per year (actual production of converted material is about 75000 t·U per year). Commercial suppliers are France (40% of world supply), Canada (26%), the Russian Federation (18%), the USA (10%) and the UK (6%). There is also some conversion performed for domestic use only in China, Argentina, India and Romania.

4.4. Enrichment of uranium

There are two processes used commercially today: Gaseous diffusion and centrifuge enrichment. The first process is used in France and the USA only, the rest of the world uses the centrifuge process.

Commercial suppliers with a total capacity of about 52×10^6 SWU per year[5] are the Russian Federation (with 39% of world market), the USA (22%), France (21%), the UK (6%), the Netherlands (5%), Germany (3%), China (2%), Japan (2%), and Brazil (0.2%).

[5] As an illustration a typical 900 MW(e) LWR needs about 20 tons of heavy metal annually. If it is enriched to 4% of ^{235}U it requires about 160 tons of natural uranium UF6 feed and the expenditure of about 10^5 SWU.

4.5. Fuel design, fabrication and operational performance

First the UO_2 fuel for LWR will be discussed, and then MOX fuel for LWR, fuel for PHWR, fuel for fast reactors, fuel for HTGR, and finally advanced fuel designs such as low strain resistant and inert matrix fuels will be shortly presented.

4.5.1. UO_2 fuel for LWR

As mentioned before, LWR fuel fabrication technology using UO_2 was established already in the late 1940s. A lot of R&D (e.g. optimization of materials, thermohydraulics, neutronics, etc.) has been continuously performed till up today resulting in a high level of economics and reliability. Energy yield increased from initially about 20 GW·d/t·U up to 60 GW·d/t·U today, enrichment from 2% to close to 5% ^{235}U, cycle length from 1 year to 2 years, and core power density from 80 kW/L to 108 kW/L. Combinations of UO_2 with burnable neutron poisons such as Gd_2O_2 are state of the art. Failure rates decreased from 10^{-3} to 2×10^{-6}; currently one of the main causes of fuel rod failures is fretting. BWR fuel assemblies started with 6 × 6 fuel rods in a square geometry and are now up to 10 × 10; similarly, fuel assemblies of PWR designed in the western world increased the number of rods from 14 × 14 up to 18 × 18. PWR fuel of WWER (Russian design) reactors use a hexagonal geometry for their fuel assemblies and increased the number of fuel rods per assembly from 126 rods (in WWER 440) to 316 rods (in WWER 1000).

Worldwide operating commercial designers and suppliers of fuel (UO_2) for (western design) PWR and BWR are Westinghouse, ENUSA and AREVA NP; additionally GE/Hitachi supplies fuel for BWR. Fuel for WWER is currently mainly supplied by TVEL (Rosatom) and partly by Westinghouse. Additionally, there are some domestic suppliers in Argentina, Brazil, China, Japan, India, and the Republic of Korea.

Worldwide there exists a UO_2 fuel fabrication capacity of about 11000 t_{HM}/a. UO_2 fuel fabrication facilities operate in Belgium (with about 6% of world capacity), Brazil (1%), China (1%), France (11%), Germany (6%), Japan (15%), the Republic of Korea (4%), the Russian Federation (14%), Spain (3%), Sweden (4%), the UK (8%), and the USA (27%).

UO_2 fuel design and production has reached a high level of maturity and will continue to dominate the supply of fuel for LWR in the future.

4.5.2. Mixed U/Pu fuel for LWR

R&D of mixed U/Pu (MOX) fuel for LWR started in the 1950s and application of MOX fuel in LWR reached commercial scale in the 1980s. For example, first MOX fuel loading into a LWR occurred in Germany as early as 1966. However, originally, MOX fuel was intended to be used in nuclear fuel cycles of fast reactors, but due to the slowdown of deployment of fast reactors in the 1980s, it was used also in LWR to decrease the volumes (and heat) of high level waste and to limit the increasing inventories of separated Pu from power reactors thereby decreasing the risk for proliferation. Typically, about 30% of a LWR core could be filled with MOX fuel assemblies. Use of MOX in thermal reactors increases the uranium utilization efficiency by a factor of two.

The main commercial designer and supplier of MOX fuel are BNFL and AREVA NP. Total capacity of MOX LWR fuel fabrication in the world is currently about 500 t_{HM} per year.

Commercial MOX fabrication facilities are located in France (with about 40% of world capacity), the UK (25%), Japan[6] (28%), and Belgium (7%). A facility for domestic supply of MOX to BWR operates also in India. In the Russian Federation, currently, Pu recycling is performed for fast reactor fuel only and in Russian thermal reactors MOX is not considered as an option; however conceptual studies are performed on how to introduce MOX fuel also into WWER reactors in the future, especially by converting weapons grade Pu. In the USA, there were 5 pilot MOX fabrication facilities in operation till 1976 that were shut down after the US government's decision to stop reprocessing of LWR fuel. In Germany, a commercial MOX fabrication facility operating from 1972 was shut down in 1991 also mainly due to political decisions.

MOX fuel is used in LWR in the following countries: Belgium, France, Germany, India, Japan, and Switzerland.

Operational LWR experience with MOX fuel has shown that it is equivalent to UO_2 fuel, i.e. about the same burnup can be achieved and no failures have occurred that were specific for MOX fuel. Future perspectives of MOX include higher burnup and LWR cores consisting of 100% MOX assemblies.

4.5.3. Fuel for PHWR

Majority[7] of operating PHWR are of CANDU design developed in Canada (AECL). The core of such a PHWR consists of horizontal D_2O cooled pressure tubes housing the fuel elements made of natural uranium surrounded by a D_2O moderator. First CANDU fuel elements had seven UO_2 fuel rods; currently, most CANDU fuel elements have 37 rods and recent designs have 43 rods, e.g. the CANFLEX design developed jointly by Canada and the Republic of Korea. A PHWR fuel element is much shorter (~ half a meter) than a LWR fuel element (several meters).

A continuous feedback from national and international improvement programmes within the CANDU user group has resulted in failure rates as low as 5×10^{-6}.

The average discharge burnup is about 8 MW·d/t (up to 10 W·d/t) which requires frequent change of fuel in the core which is achieved by online refueling.

The worldwide fabrication capacity for PHWR fuel is about 4000 t_{HM}/a. The following countries (with operating PHWR) have domestic facilities for their fuel supply: Argentina (with 4% of world capacity), Canada (64%), China (5%), India (14%), the Republic of Korea (10%) and Romania (3%).

State of the art PHWR fuel designs include a slight ^{235}U enrichment to increase the economics via increasing the power and burnup; also MOX fuel is being studied. A large R&D programme, called DUPIC, is pursued in the Republic of Korea with the goal to convert SNF from PWR into fuel for CANDU reactor (see Section 3.4).

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[6] Full capacity to be reached in 2012.

[7] There is only one operating PHWR unit with a different design (in Argentina), a pressure vessel type reactor, designed from Siemens.

4.5.4. Fuel for fast neutron reactors

MOX fuel

Fuel for operating fast reactor consists mainly of mixed oxide fuel (MOX). Typically, in fast reactor fuel the pins have stainless steel cladding and are arranged in assemblies with hexagonal geometry. The Pu content in fast reactor MOX fuel is considerably higher than in LWR MOX fuel. The burnups achieved in fast reactor are in the order of magnitude of 100 to 200 GW·d/t_{HM}.

In France, U/Pu oxide fuel for fast neutron sodium cooled reactors (RAPSODIE) was used starting in the 1960s. In the Russian Federation, the first experimental MOX fuel assemblies for BR-5, BR-10 and BOR-60 were made in the 1970s. Thereafter, from 1980 till 1992 also in BN-350 MOX assemblies were tested. Although the operating fast reactor BN-600 in the Russian Federation has a HEU oxide driver core many MOX test assemblies have been successfully irradiated since the 1990s. In Japan, since 1973, the MOX fuel for the fast reactor JOYO and MONJU has been produced domestically.

The total worldwide fabrication capacity for fast reactor MOX fuel amounts to about 200 t_{HM}/a. Semi commercial fabrication facilities for the production of MOX fuel for fast reactors exist in France (with about 80% of world fabrication capacity), the Russian Federation (17%)[8] and Japan (3%).

Metal fuel

In the USA, a long term R&D programme on metal alloy fuel (but also MC and MN fuel) was performed in the experimental fast reactor EBR-II since the 1960s till the beginning of the 1990s.

Carbide and nitride fuel

Already a long time ago mixed uranium plutonium mono-carbide (MC) and mono-nitride (MN) have been identified as candidates for liquid metal cooled fast reactor fuel due to their high thermal conductivity and excellent chemical compatibility with sodium coolant [16]. R&D programmes for metal carbide (MC) and metal nitride (MN) fuel were performed in the USA, France, Germany, the UK, the Russian Federation, Japan and India. In the Russian Federation, a uranium carbide (UC) core was in operation in the BR-5 reactor for 6 years achieving a burnup of 9 atom%. In France, as part of the FUTURIX programme, MC fuel is being tested in the PHENIX reactor. Compared to MOX fuel, however, the experience with MC and MN is still very limited.

As already mentioned earlier India is the first country to develop plutonium rich (66%) mixed carbide fuel and to successfully operate it from 1985 up to today in their experimental FBTR achieving burnups beyond 150 GW·d/t without failures.

R&D for fast reactor fuel

A lot of R&D programmes on the development of different fast reactor fuel are pursued worldwide. In the Russian Federation*,* R&D is ongoing to develop, in addition to sol-gel, ammoniac granulation, carbonaceous co-precipitation and plasma-chemical conversion processes, the commercial application of vibro-compaction technology for MOX fuel fabrication. Pre-industrial scale of the vibro-compaction process is well established. In Japan, several advanced technologies for fabrication of MOX and U/Pu metal fuel for metal cooled fast reactor are under investigation. In France, MOX fuel technology for metal cooled

[8] Russian fabrication facility is currently under construction.

(sodium) fast reactors is considered practically mature. Within the ongoing French development programme for gas cooled thermal and fast reactors the coated particle fuel concept is studied. Within the Generation IV and GNEP programme the USA is investigating several advanced fuel designs for fast reactors using also the advanced test reactor ATR at INEL.

4.5.5. Fuel for high temperature gas cooled reactors

Fuel for high temperature gas cooled reactors (HTGR) was developed and used up till now in demonstration plants only. Two concepts were realized: the pebble bed design with coated particles in a ball shaped graphite shell (e.g. AVR in Germany), and the prismatic graphite block design (e.g. FSV in the USA) filled with coated particles.

Both designs are pursued today in ongoing development efforts, e.g. in France within their gas cooled fast reactor (GFR) programme, in the Republic of Korea and in the USA studying TRISO fuel with ZrC coating and within the VHTR programme of the Generation IV initiative.

Powder agglomeration processes or wet-chemical processes (sol-gel) for the gelation of droplets from a solution containing (thorium and) uranium could be used to produce the kernels (coated particles).

The chances for HTGR fuel depend on the introduction of commercially used HTGRs. Closest to this situation today is the South-African concept of a pebble bed reactor. Other countries like China are also interested in this development.

4.5.6. Future advanced fuel designs

There are two fuel types being studied internationally with an interesting potential for future technological applications in thermal and fast neutron reactors:

- Low Strain Resistant Fuel (LSRF);
- Inert Matrix Fuel (IMF).

The basic (old) idea of Low Strain Resistant Fuel (LSRF) is to add dopants to UO_2 to decrease the creep resistance of UO_2 and thus minimizing mechanical interaction between the fuel pellets and the cladding tube (PCI) primarily during power ramping, but also because of fuel swelling, both effects limiting the allowable duty of fuel. Recent development programmes including in pile experiments in the Russian Federation adding oxides of Al, Si and Nb showed promising results, i.e. a considerable decrease of mechanical PCI, enabling a stronger degree of power ramping during operation and providing a potential for higher burnups.

The basic idea of Inert Matrix Fuel (IMF) is to use composite fuel with the fissile material dispersed in a metallic (or ceramic) matrix with high thermal conductivity resulting in low fuel operating temperatures and consequently in low fission gas release. The IMF concept can be realized in homogenous solid solutions such as oxide, nitride or metal, or in heterogeneous materials such as cercer, cermet and metmet. This so-called ‘cold’ fuel might allow raising the burnup of LWR fuel up to 120 GW·d/t, and increase safety margins during normal operation, transients and accidents.

CHAPTER 5
BACK END OF THE NUCLEAR FUEL CYCLE

5.1. Management of spent nuclear fuel

A typical 1000 MW(e) LWR produces about 200 to 350 m^3 of low[9] and intermediate[10] level waste, and about 20 m^3 (30 t_{HM}) of spent nuclear fuel (SNF) per year[11]. The currently worldwide existing reactor fleet produces about 10000 t_{HM} of SNF per year[12] with 35% each generated in Western Europe and America, and 15% each in Eastern Europe and Asia. The accumulated SNF in 2005 was about 280000 t_{HM} of which about 30% had been reprocessed; in 2020 the total amount of SNF is projected to reach about 440000 t_{HM}.

In principle there are two ways to manage spent nuclear fuel:

- reprocessing and recycling of U/Pu in the SNF (and possibly of some minor actinides and fission products) and disposing of the rest (also called a closed nuclear fuel cycle);
- direct disposal of the SNF (also called an open nuclear fuel cycle).

In both cases, the material to be finally disposed is high level waste[13] and in the case of reprocessing long lived low and intermediate level waste (LLILW), both requiring geological disposal.

The strategy of SNF management is currently very different in nuclear countries [17]. Some countries have decided to directly dispose of their SNF; examples are Finland, Sweden, Canada and Spain. Other countries have chosen to use at least partially reprocessing and recycling of their SNF; examples are France, Japan, Switzerland, the Russian Federation and the UK. Belgium, Germany, and the Netherlands, after practicing reprocessing and recycling till the end of the 1990s, decided to directly dispose of SNF in the future. The USA, on the other hand, after a long period with development of a closed fuel cycle, starting in the 1970s changed their policy to direct disposal of SNF; but recently the USA is considering to reinstall reprocessing and recycling of their SNF (as part of their GNEP programme). The majority of nuclear countries (mostly with a small nuclear power programme) are practising a wait and see strategy in regard to management of SNF.

Thus, currently, intermediate storage of high level waste (HLW) is being practised in all nuclear countries. For SNF there are two technologies available for intermediate storage:

- wet storage of SNF in pools mostly at reactor site;
- dry storage of SNF in casks cooled by ventilation or natural air circulation at reactor site or in dedicated facilities.

Typically, on-site wet storage of SNF is done for about the first 10 years after discharge. Originally long term interim storage facilities were licensed for about 50 years. In the meantime, operation of such facilities is considered for as long as 100 years to gain time for

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[9] Low level waste (LLW) consists of tools, clothing, filters, etc., with small amounts of short lived radioactivity. Most of it is suitable for shallow land burial.

[10] Intermediate level waste (ILW) contains higher amount of radioactivity and may require shielding. It consists of resins, chemical sludges, contaminated materials, etc. It is solidified in concrete or bitumen for disposal.

[11] Compared to a coal fired plant with the same electrical output generating 400000 tons of ash per year this is a very small amount.

[12] This amount of nuclear waste (10000 tons) produced per year is to be compared to 280 million tons of ash produced per year by the existing fleet of coal fired power plants.

[13] High level waste (HLW) is highly radioactive and hot, i.e. it requires shielding and cooling.

selection of the final depository and to keep the option for recycling of SNF open. Studies on long term dry storage revealed the need for R&D programmes investigating several long term effects, e.g. interaction and creep behaviour of SNF materials.

Generally, there is international agreement to finally dispose of high level waste in deep geological repositories with stable structures, e.g. granite, crystalline rock, clay, salt, etc. However, to date there is no practical urgent need for final high level waste deep geological depositories, as surface storage of SNF for 40 to 50 years instead of 10 years reduces heat and radioactivity of SNF by a factor of about two which makes handling and storage much easier compared to SNF directly unloaded from the reactor.

The process of selecting appropriate deep geological repositories for final deposal of high level waste (SNF or waste from reprocessing) is currently underway in several countries with the first (probably in Sweden, Finland or the USA) expected to be commissioned around 2020. Most countries with nuclear power have a specific organisation responsible for the development of geological disposal facilities. In parallel to national solutions of final disposal of HLW also multinational concepts are discussed but so far with limited progress.

An example of a national solution for a final repository of SNF is the Yucca Mountain site in the USA that has been studied since 1978; licence application was submitted in 2008. In Finland also the site for a final depository has been selected and a facility for underground rock characterization (called ONKALO) is under construction since 2004.

5.2. Reprocessing and recycling of spent nuclear fuel

There is considerably broad experience available for reprocessing U/Pu oxide fuel since many decades. For other types of fuel the experience is very limited with the exception for metallic fuel.

Similar to enrichment, reprocessing is sensitive in regard to proliferation; additionally, it is also sensitive to environmental protection (chemical and radio toxicity). Thus, only in a few countries, namely the USA[14], France and the UK, commercial reprocessing was performed up till now; foreign customers of these commercial facilities were Belgium, Germany, the Netherlands, Japan and Switzerland.

Worldwide capacity of reprocessing of spent nuclear fuel (SNF) amounts to about 5600 t_{HM} per year with an accumulated experience of about 80000 t_{HM} in over 50 years. Reprocessing facilities are located in the UK (at Sellafield, with 43% of world capacity)[15], France (La Hague, 30%), Japan (Rokkasho, 14%)[16], the Russian Federation (Mayak, 7%), India (Tarapur and Kalpakkam, 5%), and China (LanZhou, 1%)[17].

Currently, the most developed and widely used process in industrial applications is called PUREX (Plutonium and Uranium Recovery by Extraction) originally developed by ORNL in the USA already in the late 1940s. This aqueous process can be applied to UO_2, MOX from LWR and fast reactors, and metallic fuel and separates uranium and plutonium from the minor actinides and fission products. Its disadvantage (in regard to proliferation) is the pure Pu separation and (in regard to transmutation) the inability to separate minor actinides

[14] In the USA reprocessing was stopped in the 1970s. Based on the AFCI and the recent (2006) GNEP programme the USA is reconsidering the use of reprocessing of SNF.

[15] Includes facilities for LWR and GCR fuel.

[16] Full capacity to be reached in about 2012.

[17] Under construction as of 2008.

(MA, i.e. Neptunium, Curium, Americium) from fission products (mainly ^{137}Cs, ^{90}Sr, ^{99}Tc, ^{14}C, ^{129}I).

Based on PUREX many new technologies are currently developed, such as COEX (France), NUEX (UK), NEXT (Japan), and REPA (Russian Federation). Innovative new technologies under investigation using aqueous processes include DIAMEX-SANEX (France), UREX (USA), PARC (Japan), and GANEX (France). All this processes intend to take care of the above mentioned disadvantages of the original PUREX process, i.e. most of them offer the possibility to separate individual minor actinides (MA) and fission products and do not produce pure Pu.

The non-aqueous process for reprocessing is commonly called pyro-electrolytic processing. This process has been studied in the past (e.g. at ANL USA for metallic fuel) and is currently investigated again in several countries such as the Russian Federation, France, Japan, and the USA for different kinds of fuel (including oxides and nitrides). Another non-aqueous process is the volatile and reductive extraction process, which for example is being studied in Japan (called FLUOREX).

5.3. Partitioning and transmutation

The concept of partitioning and transmutation (P&T) is to relieve waste management of the burden caused by long living minor actinides and fission products in spent nuclear fuel (SNF) that generate most of the long term heat and radiation in high level waste. This is achieved by, firstly, separating these nuclei from SNF, and, secondly, reinserting them into a reactor where they will be destructed (transmuted) by neutron capture or fission, ending up with short lived or stable isotopes. Studies have shown that by using P&T the time needed for geological repositories to reach a radioactivity level of natural uranium ore can be reduced by several orders of magnitude. Such transmutation can be performed in thermal and fast reactors, but also in accelerator driven systems.

An accelerator driven system [18] consist typically of a subcritical reactor and co-located an external accelerator that produces high energy particles, e.g. protons that hit a heavy material target inside the reactor generating a high flux of neutrons (called spallation). Additionally, a separation unit is needed to separate transmuted elements from the reactor fuel. An example of such a system is the Rubbia concept using a cyclotron and thorium fuel. Currently, many R&D activities are performed worldwide on this system, e.g. in Europe (EIP), Japan (TEF-P), the Republic of Korea (HYPER), the Russian Federation and the USA.

CHAPTER 6
FINAL CONSIDERATIONS

This report is intended to provide a short general overview of innovative nuclear reactors and fuel cycle technologies in IAEA Member States.

It has been elaborated mainly to establish a basis within the INPRO project that could be used to explore the realistic possibilities and feasibilities to develop attractive innovative nuclear fuel cycles to commercial maturity in terms of schedules, and needed and available resources.

To achieve this goal the first step could be a collection of information on performed and ongoing R&D efforts in INPRO Member States, primarily technology holders and developers, in a suitable format using all sources available including IAEA databases.

The next step could be an evaluation of the information gathered and distribution of the results to all interested INPRO Member States. This should enable developing countries of the INPRO Member States to co-ordinate their resource limited R&D efforts with ongoing R&D activities in technology holder countries. Another opportunity of this activity could be for developing countries to get involved in multinational nuclear fuel centre activities thereby solving some sensitive issues of front and back end of their nuclear fuel cycle.

The latest development of the INPRO programme can be downloaded from the IAEA web site as follows: www.iaea.org/INPRO

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Detailed references to each individual section are given in the full report attached on a CD-ROM.

ABBREVIATIONS

ABR	Advanced burner reactor
ACR	Advanced CANDU reactor
ADS	Accelerator Driven System
AECL	Atomic Energy Canada Limited
AFCI	Advanced Fuel Cycle Initiative
AGR	Advanced gas reactor
AHWR	Advanced heavy water cooled reactor
ANL	Argonne National Laboratory (USA)
ATR	Advanced test reactor (USA)
BWR	Boiling water reactor
CCFR	China commercial fast reactor
CEFR	China experimental fast reactor
CPFR	China prototype fast reactor
DFR	Dounreay fast reactor
EBR	Experimental breeder reactor
EFFBR	Enrico Fermi fast breeder reactor
FBR	Fast breeder reactor
FBTR	Fast breeder test reactor
FCF	Fuel cycle facility
FSV	Fort St. Vrain (US nuclear plant)
FR	Fast reactor
GCR	Gas cooled reactor
GE	General Electric (US company)
GFR	Gas cooled fast reactor
GHG	Greenhouse gas
GIF	Generation IV International Forum
HEU	Highly enriched uranium
HLW	High level waste
HTGR	High temperature gas cooled reactor
HWR	Heavy water cooled reactor
I&C	Instrumentation and control
ICG	International Coordinating Group in INPRO
ICRP	International Commission on Radiological Protection

IIASA	International Institute for Applied System Analysis
IMF	Inert matrix fuel
INEL	Idaho National Energy Laboratory (USA)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
LEU	Low enriched uranium
LFR	Lead cooled fast reactor
LSRF	Low strain resistant fuel
LWR	Light water reactor
MA	Minor actinides (i.e. Cm, Am, Np)
MC	Metal carbide (fuel)
MHR	Modular helium cooled reactor
MN	Metal nitride (fuel)
MNFC	Multilateral fuel cycle (INPRO)
MOX	Mixed (U, Pu) oxide fuel
MSR	Molten salt reactor
NM	Nuclear material
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission (USA)
ORNL	Oak Ridge National Laboratory (USA)
PCI	Pellet Cladding Interaction
PBMR	Pebble bed modular (gas cooled) reactor
P&T	Partitioning and transmutation
PFR	Prototype fast reactor
PFBR	Prototype fast breeder reactor
PHWR	Pressurized heavy water reactor
PRIS	Power Reactor Information System (IAEA)
PSA	Probabilistic safety analysis
PUREX	Plutonium uranium extraction process
PWR	Pressurized water reactor
RBMK	Graphite moderated fuel channel reactor
R&D	Research and development
RD&D	Research, development and demonstration
RG	Reactor grade
SCWR	Supercritical water cooled reactor
SFR	Sodium cooled fast reactor

SNF	Spent nuclear fuel
UC	Uranium carbide (fuel)
UREX	Simplified version of PUREX
VHTR	Very high temperature (gas cooled) reactor
WCR	Water cooled reactor
ZrC	Zirconium carbide

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