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Status and Trends of Nuclear Technologies

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

(Companion CD-ROM)

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FOREWORD

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was launched in the year 2000, based on resolutions 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.

This IAEA publication is part of Phase 1 of INPRO. It intends to provide an overview on the history, present situation and future perspectives of nuclear reactors and 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.

The publication was prepared 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 expert 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.

Phase 1 of the INPRO project was implemented under the IAEA Project Manager Y.A. Sokolov and the Project Coordinators, A. Omoto, A. Rao, J. Kupitz, I. Facer and C. Ganguly of the IAEA Department of Nuclear Energy. As of August 2008, INPRO has 28 members (including the EC) supporting the project.

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

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.

EDITORIAL NOTE

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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 K2orea, 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. Short history of nuclear energy

Since the startup of Atoms for Peace on 8 December 1953[1] nuclear energy production can look back at about 50 years of successful technical and economic development. Today,

[1] US president Eisenhower to the United Nations: "The United States pledges before you — and therefore before the world — its determination to help solve the fearful atomic dilemma — to devote its entire heart and mind to find the way by which the miraculous inventiveness of man shall not be dedicated to his death, but consecrated to his life".

nuclear energy contributes about 8% primary and 15% electrical energy to the global demand. Also, with regard to the impact on the environment the overall record of nuclear energy is definitively positive.

Nevertheless, there is a burden on nuclear energy: public and/or political acceptance. Public resistance against nuclear energy developed slowly, i.e. it was not there from the very beginning of the peaceful application of nuclear energy. However, it flared up considerable due to the Three Mile Island accident in the USA with negligible impact outside the containment, and even more due to the Chernobyl accident in the former USSR with its significant regional area contamination and international radioactive fallout. It is well known that public resistance against the peaceful application of nuclear energy is remarkably different from one country to another, even where the political systems are comparable.

After its start, and a rapid growth in the 1960s and 1970s, mainly problems of acceptance, but also other concerns like safety of nuclear power plants, led to a slowdown of the growth of nuclear energy application worldwide. Again 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 use of nuclear energy in Asia 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.

1.3. Potential of nuclear energy

The growing concerns on the regional and worldwide development of the climate change and many already existing environmental problems very urgently ask for energy technologies with minimized impacts on the environment. Basically, nuclear energy has a large potential to contribute to the protection of the environment as it is one of the energy sources that is not emitting greenhouse gases.

The technical potential of nuclear energy, currently, is far from being fully used. Practically, during the last 50 years only electricity has been produced by nuclear power and exclusively in large electrical nets. The enormous potential of nuclear energy for non-electrical applications, e.g. for desalination and other industrial processes such as hydrogen production, coal gasification and liquefaction, and (fossil) fuel upgrading or for heating homes or greenhouses etc., was generally neglected up till now or development started was not finished enabling commercial use.

Nuclear energy has shown that it can provide important contributions to other industries, in particular to the development of improved safety cultures, for example in the area of industrial safety analysis and practices, in electronic process control and in many other areas. Therefore, it is reasonable to analyze the present situation on the potential for further development of nuclear power into the next decades. This analysis has to look backwards and forwards in time. Looking backwards, of course, not only the successes but also the lessons learned have to be considered. Assessment of history and the current situation of nuclear technology with regard to possible future developments has to cover several aspects:

- Economics, i.e. competitiveness with other energy systems.
- Proliferation resistance, i.e. making it impossible to abuse the peaceful application of this technology, e.g. for 'regular' military purposes or for terrorist activities.
- Protection of the environment, not only against harmful radioactivity but also against unreasonable or insufficient use of resources.

- Safety, i.e. to use all possible technical means, if appropriate, preferably passive ones, to make accidents practically impossible that lead to dangerous consequences to any kind of living beings, or at least provide sufficient protection from those consequences by adequate barriers. The so-called 'defence in depth' has to be a mandatory concept to achieve this level of safety.
- Sustainability is also an important issue. Energy remains a strategic commodity, and ensuring its continuous and stable availability is one important aspect of governments' ultimate responsibility for national security and economic growth. National circumstances and policies will determine the mix of fuels necessary to contribute to the world's collective energy security, and to address the challenge of achieving sustainable development.

With regard to future perspectives of NFC services it has to be noted that, currently, such services are no longer exclusively performed in Europe (OECD countries). Twelve countries outside of OECD have built fuel cycle facilities, and many steps of the total fuel cycle can be found under development or in operation in one or more of these countries.

In Annex B of this report explicit consideration is given to the topic of regional or multilateral fuel cycle centres. As a matter of fact, this idea is very old and the history of many attempts into this direction is shortly sketched in this chapter. As attractive and basically reasonable this idea looks, it is a fact that during more than 50 years of nuclear power development not one of the raised and discussed ideas could be realized. However, it seems that now the situation becomes more favourable. On the one hand, the international climate for technical cooperation, particularly for large innovative projects, has improved significantly during the last 10 to 15 years. On the other hand, a very rapid process of commercial globalization is underway, driven, inter alia, by a rapid development of the communication technology. Also, the international political development towards increased radicalism and terrorism clearly calls for strong international cooperation particularly in such a sensitive area as the application of nuclear energy technology.

1.4. 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 (addressing all reactor types) are laid out.

Chapter 3 covers the same issues as Chapter 2 (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.

Annex A presents a Russian study on a sustainable global nuclear energy system.

Annex B discusses the option of multilateral fuel cycle centres.

CHAPTER 2.
HISTORY, CURRENT STATUS AND PROSPECTS OF NUCLEAR POWER

In the following sections the history of nuclear power is sketched from its very beginning mentioning the milestones of its development.

2.1. Short history of start of nuclear power

The most important milestones of the development of the scientific basis of nuclear power technology can be characterized as follows:

In 1934, E. Fermi proposed the ß-decay theory, combining previous work on radiation theory with Pauli's idea of the neutrino. Following the discovery by Curie and Joliot of artificial radioactivity (1934), he demonstrated that nuclear transformation occurs in almost every element subjected to neutron bombardment. This work resulted in the discovery of slow neutrons that same year.

In 1938, O. Hahn, with the support of L. Meitner and F. Straßmann experimentally proved the fission by neutrons of the heavy element uranium into much lighter fission products.

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. US Congress created the Atomic Energy Commission (AEC) in 1946. The AEC authorized the construction of the Experimental Breeder Reactor I (EBR-I) at a site in Idaho National Laboratory. The reactor generated in the laboratory the first electricity from nuclear energy on 20 December 1951. In 1950, president Truman started first political activities of international importance on the peaceful use of nuclear energy. In 1953, president Eisenhower proposed his Atoms for Peace programme, which set the course for civil nuclear energy development in the western world.

Also in the former Soviet Union (USSR), work was started at various development centres to refine existing and develop new reactor designs for power generation. The existing (military) graphite-moderated channel-type plutonium production reactor was modified for heat and electricity generation, and in 1954 the world's first nuclear powered electricity generator at the Institute of Physics and Power Engineering (IPPE) began to supply power to the (then closed) city of Obninsk.

In the mid-1950s major efforts were undertaken in the USA with the goal to show that nuclear energy could produce electricity for commercial use.

However, the beginning of this development still was under military control, in the USA as well as in the former USSR. The main USA effort was under the leadership of Admiral H. Rickover, which developed the pressurized water reactor (PWR) for military naval (particularly submarine) use. The PWR developed used enriched uranium oxide fuel and was moderated and cooled by ordinary (light) water. The Mark 1 prototype naval reactor started up in March 1953 in Idaho, and the first nuclear-powered submarine, USS Nautilus, was launched in 1954. In 1959, both the USA and the former USSR launched their first nuclear-powered surface vessels.

In order to encourage the US private industry to take over the development of commercial nuclear power plants, the AEC sponsored a pilot project with the utility Duquesne Light Co.

in Shippingport, Pennsylvania to build a 60 MW plant, which opened in 1957 as a demonstration programme. The plant was designed by Westinghouse and was based on a PWR originally developed by Admiral H. Rickover's navy programme for submarine propulsion. It operated until 1982. However, the private industry in the USA hesitated to launch fully commercial plants until the early 1960s.

In the following text the early history of thermal neutron power reactors (water or gas cooled) in specific countries will be described[2].

The first fully commercial power plant, modelled after the BORAX III experimental reactor (developed by the Argon National Laboratory, ANL, USA) was ordered in 1955 and built in 1959 for Commonwealth Edison in Morris, Illinois, USA. The plant, known as Dresden Unit One was a 200 MW, dual cycle boiling water reactor (BWR) and started up in 1960.

The break through for nuclear power in the USA occurred in the mid-1960s (mainly in 1966–67), when a kind of 'gold rush' of orders for BWRs to General Electric (GE) and PWRs to Westinghouse (W) began and continued for some years. Both reactor types used enriched uranium as the fuel in the form of ceramic UO_2 canned in metallic cladding tubes, and light water for moderation and cooling of the core. After first attempts with stainless steel as cladding tubes, very early Zircaloy-2 for BWR fuel and Zircaloy-4 for PWR fuel became internationally the standard cladding material in the western countries.

Since in the western hemisphere the USA had a virtual monopoly on uranium enrichment, several countries originally started nuclear power development on the basis of using natural uranium.

The UK took a different track than the USA that resulted in a series of reactors fuelled by natural uranium metal, moderated by graphite, and gas cooled (CO_2). The first of these 50 MW(e) Magnox types (Calder Hall-1) started up in 1956 and was still running till 2004. However, after 1963 (and 26 nuclear units in operation) no more Magnox reactors were built (and several have been shut down up to now). The UK next continued with the deployment (1976) of the advanced gas cooled reactor (AGR) units using enriched oxide fuel instead of metal fuel, before in the early 1990s conceding to the pragmatic virtues of the PWR design (Westinghouse) and deploying a 1200 MW(e) at Sizewell. Currently, worldwide there are no gas cooled power reactors operating outside the UK.

France started out with a gas–graphite design similar to Magnox and the first reactor went on line in 1956 (G-1 in Marcoule). Commercial units began operation in 1963 (Chinon A-1). The latest unit with this type of reactor was shut down in the middle of the 1990s. Later (starting towards the end of the 1960s) France settled on three successive generations of standardized PWRs, based originally on a licence agreement with Westinghouse.

Canadian power reactor development headed down a quite different track using natural uranium fuel and heavy water as a moderator and coolant. The first unit started up in 1962. The CANDU (Canadian Deuterium Uranium) design continues to be refined up to today.

In Germany, the very first reactor development also started with natural uranium fuel and heavy water as a moderator. A demonstration reactor called MZFR was built in the research centre of Karlsruhe and operated from 1965 to 1984. Additionally, a heavy water moderated and gas cooled demonstration plant was constructed in Niederaichbach (operating 1973 to

[2] The history of high temperature gas cooled reactors and of fast neutron reactors will be discussed separately in Sections 2.4 and 2.5, respectively.

1974). But rather early a basic strategic turn was taken by the industry, to follow the USA development towards nuclear power plants with BWRs and PWRs. The first commercial nuclear demonstration plant was a BWR with 15 MW(e), started in 1961 at Kahl/Germany, based on a licence of General Electric in the USA, followed by a PWR at Obrigheim shortly thereafter.

In 1964, in the Russian Federation (the former Soviet Union, USSR) the first two nuclear power plants were commissioned. A 100 MW boiling water graphite channel reactor began operating in Beloyarsk (Ural). In Novovoronezh (Wolga region), a new design, a small (210 MW) PWR known as a WWER (voda-vodyanoi energetichesky reaktor = water cooled power reactor) was built. The first large light water cooled and graphite moderated reactor called RBMK (1000 MW(e) — high power channel reactor) started up at Sosnovy Bor near St. Petersburg in 1973 and the same year saw the commissioning of the first of four small (12 MW) boiling water channel-type units in the eastern Arctic town of Bilibino for the production of both power and heat. In the Arctic north-west a slightly bigger WWER with a rated capacity of 440 MW began operating and this became a standard design in these times.

In India, the first commercial nuclear power plant (NPP) was set up at Tarapur, near Mumbai with two BWR units (TAPS 1&2). These two BWRs of 160 MW(e) capacity each are in commercial operation since October 1969 and use enriched uranium as fuel. The twelve other operating commercial NPPs in India are of the pressurised heavy water reactor of 220 MW(e) (PHWR 220) type. The first PHWR was set up at Rajasthan Atomic Power Station (RAPS). This unit, namely RAPS-1 went into commercial operation in December 1973.

In Japan, construction of the Japan Power Demonstration Reactor (JPDR), which was a BWR with 12.5 MW(e) was initiated in 1960 by a cooperation of GE and Hitachi in 1960 and was put into operation in 1963. The first commercial nuclear power plant, Tokai-1 of the Japan Atomic Power Co. Ltd (JAPCO), a Calder Hall type gas cooled reactor, was completed in 1966. After this reactor, only light water reactors, either BWRs or PWRs, have been constructed by JAPCO for nine electric power utility companies. In 1970, three nuclear power plants, Tsuruga-1 (BWR) of JAPCO, Fukushima-1 (BWR) of Tokyo Electric Power Co. Ltd (TEPCO) and Mihama-1 (PWR) of Kansai Electric Power Co. Ltd (KEPCO) were completed and synchronized with the grid.

In the 1950s and 1960s, there were also several other thermal neutron reactor concepts under investigation, some of them looking rather exotic at that time like a molten salt reactor with various core design concepts. None of them were ever built as commercial reactors in these years.

The 1970s and 1980s, were the two decades when the use of nuclear power for generation of electricity was rapidly spreading internationally, in Europe (mainly in Belgium, France, Germany, Italy, Spain, Sweden, Switzerland, and the UK), in Middle and South America (in Argentina, Brazil and Mexico), and also in the Far East, first in Japan and later also in the Republic of Korea, and also in India and China. In most cases, the LWR types BWR and PWR were built, but also HWRs, e.g. in Argentina (Canadian and German concepts) and in India and in Romania (Canadian concepts). In this time, the reactor sizes, in particular of the LWRs increased stepwise to 600, 900 and finally to 1100–1200 MW(e) units. This increase was mainly driven by economic reasons, because in parallel to a continuous improvement of the reactor technology also the construction cost (and time) was increasing in most countries partly due to increasing regulatory demands.

The cost advantage in comparison to other energy sources was most pronounced where nuclear power plants could be operated in base load. In most cases, it was possible to keep the power production cost below or at least close to other (conventional) energy systems, because

in parallel with the technological development of reactors and nuclear fuel, also the reliability, expressed in availability of the plant for production increased continuously. Already in the 1980s, some plants exceeded an availability factor of 90% and today this is the standard to be applied to every well-managed nuclear power station in the world.

Basically, the above-discussed improvement of economics and reliability during the 1970s and 1980s was achieved also for the Russian LWR reactor designs, i.e. the WWER-440 and later the WWER-1000.

2.2. Current status of nuclear power

The current worldwide situation of nuclear power (based on Refs [5–7]) is laid out in the following[3] section.

2.2.1. Current status of nuclear power worldwide

During about the last ten years up to now the total number of operating nuclear power plants is almost stagnant, i.e. around 440. By the end of 2007, this number was 435 in 31 countries. The nuclear reactors are currently producing approximately 16% of the world's electricity.

Two new nuclear power stations have been connected to the grid in 2006 (one in India and one in China), and 7 are expected to be connected in 2007 (four in India, and one each in China, the Islamic Republic of Iran and Romania); this compares to two in 2003, five in 2004, and four in 2005. There were eight nuclear power plant retirements in 2006, compared to two in 2005, five in 2004, and six in 2003.

Figure 2.1 below shows the geographical locations of nuclear power plants in the world.

FIG. 2.1. Geographical locations of NPP in the world.

The power plants are concentrated in Western Europe and the eastern part of the USA. Table 2.1 provides a worldwide overview of nuclear reactors in operation and under construction (as of end of 2006).

[3] The status of nuclear power programmes in specific countries is presented thereafter shortly in Sections 2.2.2 to 2.2.10.

TABLE 2.1. NUCLEAR POWER REACTORS IN OPERATION AND UNDER CONSTRUCTION IN THE WORLD (TABLE FROM PRIS 2006)

Country	Reactors in Operation		Long-term Shut Down Reactors		Reactors under Construction		Nuclear Electricity Supplied in 2006	
	No of Units	Total MW(e)	No of Units	Total MW(e)	No of Units	Total MW(e)	TW(e).h	% of Total
ARGENTINA	2	935			1	692	7.15	6.93
ARMENIA	1	376					2.42	41.95
BELGIUM	7	5824					44.31	54.43
BRAZIL	2	1901					12.98	3.31
BULGARIA	2	1906			2	1906	18.15	43.64
CANADA	18	12610	4	2658			92.43	15.81
CHINA	10	7572			4	3610	51.81	1.93
CZECH R.	6	3523					24.50	31.48
FINLAND	4	2696			1	1600	22.00	27.99
FRANCE	59	63260					429.82	78.07
GERMANY	17	20339					158.71	31.82
HUNGARY	4	1755					12.51	37.70
INDIA	16	3577			7	3112	15.59	2.62
IRAN					1	915		
JAPAN	55	47587	1	246	1	866	291.54	29.97
KOREA RP	20	17454			1	960	141.18	38.64
LITHUANIA	1	1185					7.94	72.30
MEXICO	2	1360					10.40	4.86
NETHERLANDS	1	482					3.27	3.47
PAKISTAN	2	425			1	300	2.55	2.74

TABLE 2.1. NUCLEAR POWER REACTORS IN OPERATION AND UNDER CONSTRUCTION IN THE WORLD (TABLE FROM PRIS 2006) (cont.)

Country	Reactors in Operation		Long-term Shut Down Reactors		Reactors under Construction		Nuclear Electricity Supplied in 2006	
	No of Units	Total MW(e)	No of Units	Total MW(e)	No of Units	Total MW(e)	TW(e).h	% of Total
ROMANIA	1	655			1	655	5.18	9.00
RUSSIA	31	21743			5	4525	144.64	15.91
S.AFRICA	2	1800					10.07	4.41
SLOVAKIA	5	2034					16.60	57.16
SLOVENIA	1	666					5.29	40.26
SPAIN	8	7450					57.43	19.82
SWEDEN	10	9097					65.05	47.98
SWITZERLAND	5	3220					26.37	37.41
UK	19	10965					69.39	18.40
UKRAINE	15	13107			2	1900	84.91	47.53
USA	103	99257	1	1065			788.31	19.42
Total	435	369682	6	3879	29	23641	2660.86	18.76

Note: The total includes the following data in Taiwan, China:

— 6 units, 4921 MW(e) in operation; 2 units, 2600 MW(e) under construction;

— 38.32 TW(e).h of nuclear electricity generation, representing 19.49% of the total electricity generated there;

The total share is related only to the countries with NPPs in operation.

In 2007, Browns Ferry 1 of TVA in the USA was restarted based on an extensive upgrading project.

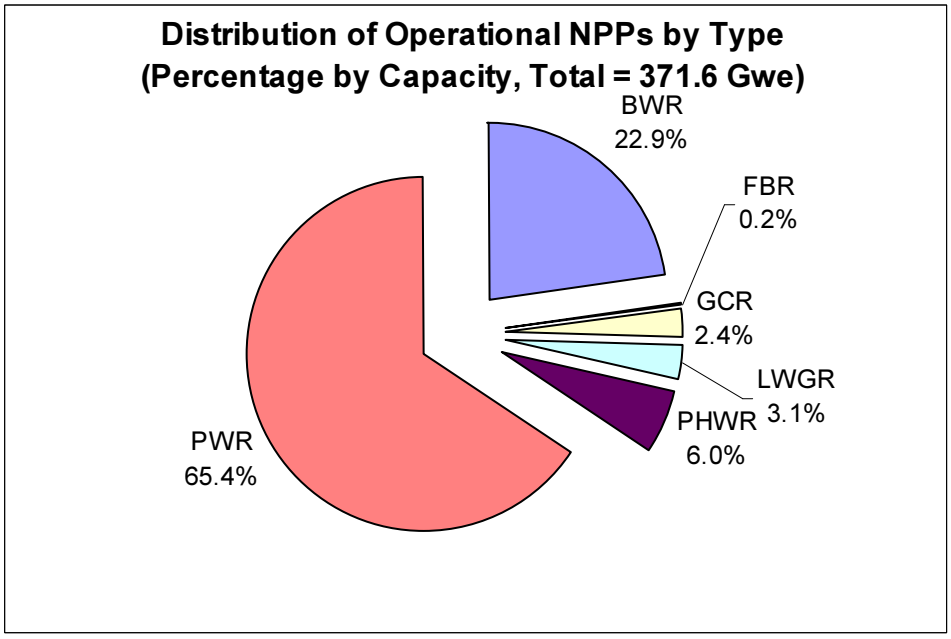


FIG. 2.2 Types of nuclear reactors in operation (2007).

Figure 2.2 shows that a big majority (~88%) of the above-referred nuclear capacity belongs to power plants with light water cooled reactors (PWR, BWR and WWER). More than 40 plants with together about 25 GW(e) (gross) are D_2O moderated (i.e. HWRs), most of them of the CANDU type; this corresponds to 6% of the nuclear world capacity. In the UK, there are 9 GW(e) installed in 18 plants of the gas cooled types GGR and AGR (i.e. the gas cooled Magnox type and the advanced gas cooled reactors); this is about 2% of the nuclear world capacity. Finally, there are 15 Russian RBMK (or LWGR) plants that cover about 10 GW(e) (gross) corresponding to 3%. There is a single RBMK running outside the Russian Federation, i.e. in Lithuania.

Nuclear power (i.e. electricity) production being competitive achieved a high level internationally, but probable not as high as theoretically possible as this level was influenced not only by technological factors. Besides influences of market prices for labour and material there was and there is a strong political influence, which varies from country to country. One politically caused fact is the liberalization of the market for electrical power, which started first in North America and now also proceeds in Europe. In principle, it can be stated that the nuclear power production has been and also will be competitive in any reasonable energy mix policy.

The reliability of commercial nuclear power plants has improved dramatically during the last decade. While the world average nuclear power plant availability in 1991 was 74%, by the end of 2001 it had improved to over 83% and was around 85% in 2002 [8, 10].

The safety of nuclear power stations has achieved a very high level internationally. In fact, since the severe accident at Chernobyl there was no accident in a nuclear power station that caused serious harm to the people in and outside the plant, and to the environment. In principle, it may be stated here that the power reactor technology in the past was always able to follow the demands of increasing safety requirements and there is no reasonable argument visible why it should not be able to do so in the future.

In the following Sections 2.2.2 to 2.2.9 the current status (2005) of nuclear power in selected countries (Canada, China, France, India, Japan, the Republic of Korea,

the Russian Federation, and the USA) is laid out in some detail [11]. In Section 2.2.10 a short summary of the situation in other countries with operating nuclear power plants is presented.

2.2.2. Current status of nuclear power in Canada

As of end of 2006, 18 reactors are in operation in Canada producing about 15% of the electricity. In 1997, after a performance assessment eight CANDU reactors were shut down for refurbishment. Till 2005, four of these units were restarted. The remaining four units are planned to be restarted in the future too. In one Canadian state (Alberta) plans have been announced during 2007 to order a new nuclear power plant that should be used to produce heat needed to extract oil from oil sands.

2.2.3. Current status of nuclear power in China

Moves to build nuclear power commenced in mainland China in 1970 and the nuclear industry has now moved to a steady development phase. Technology transfer has been received originally from France, Canada and the Russian Federation, with national development of PWRs based largely on French design. In 2010, the State Power Grid Corporation expects to supply 3810 billion kW·h (by all energy sources) with 852 GW(e) installed capacity. Growth of capacity is then expected to slow down towards 2020, when installed capacity is expected to reach 1330 GW(e).

Nuclear power has an important role in energy supply, especially in the coastal areas remote from the coalfields and where the economy is developing rapidly. In 2006, nuclear power provided 51.8 billion kW·h — 1.9% of total, and there was a capacity of 8.6 GW(e) installed. The government plans to increase nuclear generating capacity to 40 GW(e) by 2020, with an additional 18 GW(e) nuclear capacity being under construction then, requiring an average of 2 GW(e) per year being added. In May 2007, the National Development and Reform Commission announced that its target for nuclear generation capacity in 2030 was 160 GW(e).

The first two nuclear power plants in mainland China were at Daya Bay near Hong Kong (China) and Qinshan, south of Shanghai, with construction starting in the mid-1980s. Table 2.2 shows the currently operating plants.

TABLE 2.2. OPERATING NUCLEAR POWER REACTORS IN CHINA

Units	Province	Type	Net capacity (each)	Start operation*	Operator
Daya Bay-1&2	Guangdong	PWR	944 MW(e)	1994	CGNPC
Qinshan-1	Zhejiang	PWR	279 MW(e)	April 1994	CNNC
Qinshan-2&3	Zhejiang	PWR	610 MW(e)	2002, 2004	CNNC
Lingao-1&2	Guangdong	PWR	935 MW(e)	2002, 2003	CGNPC
Qinshan-4&5	Zhejiang	PHWR	665 MW(e)	2002, 2003	CNNC
Tianwan-1	Jiangsu	PWR (WWER)	1000 MW(e)	2007	CNNC
Tianwan-2	Jiangsu	PWR (WWER)	1000 MW(e)	2007	CNNC
Total (11)			8587 MW(e)		

* *dates are for start of commercial operation.*

In May 2004, the China National Nuclear Corporation (CNNC) applied to build eight (4 pairs) of large new reactors:

- Lingao phase 2 (Lingdong) in Guangdong province, to duplicate the CPR-1000 Lingao nuclear plant, based on the same French technology as phase 1;
- Qinshan phase 4 in Zhejiang province, duplicating the indigenous CNP-600 units, upgraded to 650 MW(e);
- Sanmen, in the Zhejiang province, and Yangjiang, in Guangdong province, 500 km west of Hong Kong (China), using advanced foreign technology and design.

The eleventh 5-year plan 2006–2010 has firmer environmental goals than previously, including a reduction of 20% in the amount of energy required per unit of GDP, i.e. a 4% reduction per year.

Nuclear power developments originally proposed in the eleventh 5-year plan included:

- four CNP/CPR-1000 units at Hongyanhe, Liaoning province (NE);
- two 1000 MW(e) units at Haiyang, Shandong province (now a 1100 MW(e) AP-1000);
- two 1000 MW(e) units at Hui'an, Fujian province;
- two units at Hongshiding, Shandong province;
- two units at Tianwei, Lufeng in Guangdong province;
- two units at Taishan Yaogu in Guangdong.

China Power Investment Corporation (CPI) announced that it expected to start construction of the first plants in 2006. CPI is also expected to determine where the next Generation III reactor is built.

More than 16 provinces, regions and municipalities have announced intentions to build nuclear power plants in the twelfth 5-year plan 2011–2015. These include Henan and Sichuan, as well as those listed above — most of which have a preliminary project approval by the central government but are not necessarily scheduled for construction. Provinces will put together firm proposals with reactor vendors by 2008 and submit them to the central government's National Development and Reform Commission for approval before 2010.

Plans have been announced to build a prototype pebble bed gas cooled reactor within the next 10 years.

2.2.4. Current status of nuclear power in France

Fifty-nine reactors are in operation in France and supply about 78% of its electricity. A plan was announced to start construction of a new reactor (1.6 GW(e) EPR type) at Flammanville in 2007–2008 with operation expected to begin in 2012–2013. Power uprates were approved for four of the latest designed PWRs (Type N4).

France is firmly committed to nuclear power. Figure 2.3 shows an exemplary long term scenario of the French nuclear power system.

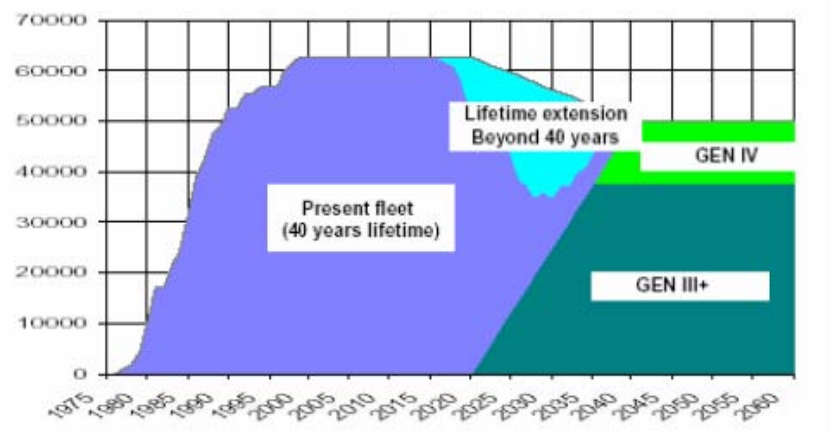


FIG. 2.3. Example of a French nuclear scenario.

The scenario shows a constant nuclear power production of the current fleet of reactors till about 2025, and thereafter a slight decrease (of about 15%) till 2040, followed by a constant supply of power. A licence extension of current plants is taken into account, Generation III reactors (see Section 2.6.1) would start to replace retired plants of the current generation in 2025, and finally after about 2040 Generation IV reactors would be added.

2.2.5. Current status of nuclear power in India

The Atomic Energy Establishment was set up at Trombay, near Mumbai, in 1957 and renamed as Bhaba Atomic Research Centre (BARC) ten years later. Plans for building the first pressurised heavy water reactor (PHWR) were finalised in 1964, and this prototype — Rawatbhata-1, which had Canada's Douglas Point reactor as a reference unit, was built as a collaborative venture between Atomic Energy of Canada Ltd and NPCIL. It started up in 1972 and was duplicated; subsequent indigenous PHWR development has been based on these units.

India's nuclear power programme proceeds largely without fuel or technological assistance from other countries. Till the mid-1990s its power reactors had some of the world's lowest capacity factors, reflecting the technical difficulties of the country's isolation, but rose impressively from 60% in 1995 to 85% in 2001–2002. India's nuclear energy self-sufficiency extends from uranium exploration and mining through fuel fabrication, heavy water production, reactor design and construction, to reprocessing and waste management. It has an operating small fast breeder reactor and is building a much larger one. It is also developing technology to utilize its abundant resources of thorium as a nuclear fuel.

Electricity demand in India has been increasing rapidly, and the 534 billion kilowatt hours produced in 2002 was almost double the 1990 output, though still representing only 505 kW·h per capita in this year. This per capita figure is expected to almost triple by 2020, with 6.3% annual growth. Coal provides over half of the electricity at present, but national reserves are limited.

Nuclear power supplied 15.6 billion kW·h (2.6%) of India's electricity in 2006 from 3.5 GW(e) (of 110 GW(e) total) capacity and this will increase steadily as new nuclear plants come on line. India's fuel situation, with shortage of fossil fuels, is driving the nuclear investment for electricity, and 25% nuclear contribution is foreseen by 2050, i.e. one hundred times the 2002 capacity. Almost as much investment is necessary in the grid system as in power plants. In 2006, about US $9 billion was committed for power projects, including 9354 MW(e) of new generating capacity, taking forward projects to 43.6 GW(e) and US $51 billion. In 2007, a report (KPMG) said that India needed to spend

US $120–150 billion on power infrastructure over the next five years, including transmission and distribution. It said that distribution losses are currently some 30–40%, worth more than US $6 billion per year.

The Nuclear Power Corporation of India Ltd (NPCIL) is responsible for design, construction, commissioning and operation of thermal nuclear power plants; it has 15 small and two mid-sized nuclear power reactors in commercial operation, six under construction — including two large ones (WWER type) and a fast breeder reactor, and more are planned.

TABLE 2.3. INDIA's OPERATING NUCLEAR POWER REACTORS

Reactor	Type	MW(e) net, each	Commercial operation	Safeguards status
Tarapur 1&2	BWR	150	1969	site-specific
Kaiga 1&2	PHWR	202	1999–2000	
Kaiga 3	PHWR	202	2007	
Kakrapar 1&2	PHWR	202	1993–95	by 2012
Kalpakkam 1&2 (MAPS)	PHWR	202	1984–86	
Narora 1&2	PHWR	202	1991–92	by 2014
Rawatbhata 1	PHWR	90	1973	site-specific
Rawatbhata 2	PHWR	187	1981	site-specific
Rawatbhata 3&4	PHWR	202	1999–2000	by 2010
Tarapur 3&4	PHWR	490	2006	
Total (17)		3779		

2.2.6. Current status of nuclear power in Japan

Japanese energy policy has been driven by considerations of energy security and the need to minimise dependence on imports. The main elements of the strategy regarding nuclear power are:

- Continue to have nuclear power as a major element of electricity production.
- Recycle uranium and plutonium from spent fuel, initially in LWRs, and start reprocessing domestically from 2005 onwards.
- Develop fast breeder reactors continuously in order to improve uranium utilisation significantly.
- Promote nuclear energy to the public, emphasising safety and non-proliferation.

In 2004, Japan's Atomic Industrial Forum released a report on the future prospects for nuclear power in the country. It integrated several considerations including a 60% reduction in carbon dioxide emissions and a 20% reduction in population (but with a constant gross domestic product). Projected nuclear generating capacity in 2050 is 90 GW(e). This would require doubling both nuclear generating capacity and nuclear share to about 60% of total power generated. In addition, some 20 GW(th) of nuclear heat could be utilised for hydrogen production. Hydrogen is expected to supply 10% of consumed energy and 70% of this could come from nuclear plants in 2050.

In July 2005, the Atomic Energy Commission reaffirmed policy directions for nuclear power in Japan, while confirming that the immediate focus would be on LWRs. The main elements are that a 30–40% share or more shall be the target of nuclear power for total generation after 2030, including replacement of current plants with advanced light water reactors. Fast

breeder reactors should be introduced commercially, but not until about 2050. Used fuel should be reprocessed domestically to recover fissile material for use in MOX fuel. Disposal of high level wastes will be addressed after 2010.

In April 2006, the Institute of Energy Economics Japan forecast for 2030 that while primary energy demand will decrease 10% electricity use will increase and nuclear share will be 41% of 63 GW(e) (total capacity estimated in 2030). Ten new units would come on line by 2030 and Tsuruga-1 would be retired.

In May 2006, the ruling Liberal Democratic Party urged the government to accelerate development of fast breeder reactors (FR), calling them "a basic national technology". It proposed an increased budget, better coordination in moving from R&D to verification and implementation, plus increased international cooperation. Japan is already playing a leading role in the Generation IV initiative, with focus on sodium cooled FR.

Since 1970, 28 BWRs (including two ABWRs) and 23 PWRs have been brought into operation.

The first ABWR (of 1315 MW(e)) were Tokyo Electric Power Co.'s (Tepco's) Kashiwazaki-Kariwa units 6 and 7 which started up in 1996–1997, and Shika-2 and Hamaoka-5 that started up in 2005. These units were built by a consortium of General Electric (USA), Toshiba and Hitachi. Two further ABWR — Ohma and Shimane-3 — are currently being commissioned or are under construction.

TABLE 2.4. JAPANESE REACTORS UNDER CONSTRUCTION

Reactor	Type	Net capacity MW(e)	Utility	Construction start	Operation*
Tomari-3	PWR	912	Hokkaido	2003	2009
Shimane-3	ABWR	1373	Chugoku	2006	2012
Ohma	ABWR	1383	EPDC	2007	2013
Total (3)		3668			

** according to latest announcements 2007 Ohma is being reviewed by regulatory authorities.*

TABLE 2.5. JAPANESE REACTORS PLANNED, ON ORDER OR INVITED

Reactor	Type	MW(e) (each)	Utility	Start * construction	Start * operation
Tsuruga 3&4	APWR	1538	JAPC	2010	2016–2017
Fukushima I - 7&8	ABWR	1380	Tepco	2009	2012–2013
Higashidori 1&2	ABWR	1385	Tepco	2008&2010	2014&2016
Kaminoseki 1&2	ABWR	1373	Chugoku	2009&2012	2014&2017
Higashidori 2	ABWR	1385	Tohoku	2013	2017
Namie-odaka	BWR	825	Tohoku	2013	2017
Sendai 3	PWR	1100	Kyushu	invited by local community	
Total (11)		14662			

** according to latest announcements 2007 Tsuruga 3-4 is being reviewed by regulatory authorities.*

After some delay due to sitting problems, units 3&4 of the Tsuruga nuclear power plant have been approved by the Fukui prefecture, and Japan Atomic Power Co has sought government approval for construction. The approval process is likely to take two years, and construction

— estimated at 770 billion yen (US $7.4 billion) — is due to start in 2010 with commercial operation in 2016–2017.

The Electric Power Development Corp has sought permission to build its Ohma nuclear plant — a 1383 MW(e) advanced boiling water reactor (ABWR) — in the Aomori prefecture. Construction started in August 2007 for commissioning in 2013. Apart from the Fugen experimental advanced thermal reactor (ATR), this would be the first Japanese reactor built to run solely on mixed oxide (MOX) fuel incorporating recycled plutonium.

2.2.7. Current status in the Republic of Korea

There is a continuing increase in electricity demand in the Republic of Korea. Economic growth and requirement for a better quality of life are the main reasons for the increased demand. It increased about seven times in 20 years since 1980 with an average annual growth rate of 10.3%; the anticipated average annual growth rate from the year 2003 to the year 2015 is estimated to reach 2.8%. Thus, the Republic of Korea must continuously increase the supply of electricity in order to meet the demand. However, the availability of conventional energy resources is very limited in the Republic of Korea: no oil, little natural gas and limited hydro power (see Fig. 2.4).

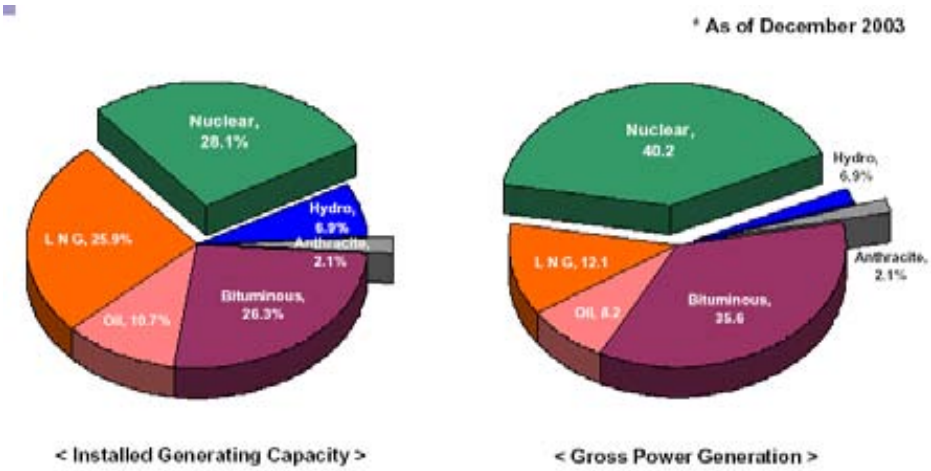


FIG. 2.4. Energy sources in the Republic of Korea.

There is coal mining, but utilization of coal is limited due to harmful environmental impacts. The role of nuclear power in electricity generation is expected to become more important in the Republic of Korea in the years to come due to increasing electricity demand and (as said above) limited national resources. The first commercial nuclear power plant Kori Unit 1 started its operation in 1978; currently there are 16 PWRs and 4 PHWRs in operation as shown in Table 2.6, and eight under construction (Table 2.4).

TABLE 2.6. POWER REACTORS OPERATING IN THE REPUBLIC OF KOREA

Reactor	Type	Net capacity	Operation
Kori 1	PWR	570 MW(e)	4/78
Kori 2	PWR	630 MW(e)	7/83
Wolsong 1	PHWR	635 MW(e)	4/83
Kori 3	PWR	950 MW(e)	9/85
Kori 4	PWR	950 MW(e)	4/86
Yonggwang 1	PWR	945 MW(e)	8/86
Yonggwang 2	PWR	945 MW(e)	6/87
Ulchin 1	PWR	950 MW(e)	9/88
Ulchin 2	PWR	950 MW(e)	9/89
Yonggwang 3	PWR (Syst 80)	989 MW(e)	12/95
Yonggwang 4	PWR (Syst 80)	989 MW(e)	3/96
Wolsong 2	PHWR	680 MW(e)	7/97
Wolsong 3	PHWR	680 MW(e)	7/98
Wolsong 4	PHWR	680 MW(e)	10/99
Ulchin 3	PWR (KSNP)	995 MW(e)	8/98
Ulchin 4	PWR (KSNP)	995 MW(e)	12/99
Yonggwang 5	PWR (KSNP)	1000 MW(e)	5/02
Yonggwang 6	PWR (KSNP)	1000 MW(e)	12/02
Ulchin 5	PWR (KSNP)	1000 MW(e)	7/04
Ulchin 6	PWR (KSNP)	1000 MW(e)	8/05
Total: 20		17533 MW(e)	

Net capacities updated from NEI August 2006.

In 2005, the capacity factor for Korean power reactors averaged 96.5%, i.e. one of the highest in the world.

TABLE 2.7. POWER REACTORS UNDER CONSTRUCTION OR ON ORDER IN THE REPUBLIC OF KOREA

Reactor	Type	Net capacity	Startup*
Shin Kori 1	OPR-1000	950 MW(e)	2010
Shin Kori 2	OPR-1000	950 MW(e)	2011
Shin Wolsong 1	OPR-1000	950 MW(e)	2011
Shin Wolsong 2	OPR-1000	950 MW(e)	2012
Shin Kori 3	APR-1400	1350 MW(e)	2013
Shin Kori 4	APR-1400	1350 MW(e)	2014
Shin Ulchin 1	APR-1400	1350 MW(e)	2015
Shin Ulchin 2	APR-1400	1350 MW(e)	2016
Total	8	9400 MW(e)	

** Latest announced commercial operation.*

KHNP and MOST are negotiating licence renewals to extend operating lifetimes by ten years, starting with Kori-1 and Wolsong-1. Power uprates of most units occurred at the end of 2005,

totalling 693 MW(e) and reflecting the fact that they had been declaring load factors of over 100% for some time.

In 2005, permits for construction of Shin Kori 1&2 and Shin Wolsong 1&2 were authorised. Some construction of Shin Kori 1&2 commenced in November 2005 and at Shin Wolsong 1&2 in June 2006. First concrete for Shin Kori 1 was poured in June 2006 and that for unit 2 is due August 2007. For Shin Wolsong first concrete is due June 2007 for unit 1 and June 2008 for unit 2. In 2005, the KSNP/KSNP+ was rebranded as OPR-1000 (optimised power reactor) apparently for Asian markets, particularly Indonesia and Vietnam. Eight operating units and four under construction are now designated OPR-1000. KHNP is hoping to interest China in the APR-1400. Construction of the first pair of third generation APR-1400 reactors — Shin Kori 3&4 — was authorised in 2006. In August 2006, KHNP placed a US $1.2 billion order with Doosan Heavy Industries for major components of these. Westinghouse has a US $300 million contract with Doosan for part of this order. KHNP expects the APR-1400 reactors to cost a total of US $5 billion (US $1850/kW$_{installed}$) and to generate power at US $0.035/kW·h. First concrete is expected to be poured in October 2008, and construction time of 51 months is envisaged.

2.2.8. Current status of nuclear power in the Russian Federation

The Russian Federation's commercial nuclear plants, with 31 operating reactors totalling 21743 MW(e), comprise:

- four first generation WWER-440/230 or similar pressurised water reactors;
- two second generation WWER-440/213 pressurised water reactors;
- nine third generation WWER-1000 pressurised water reactors with a full containment structure;
- eleven RBMK light water graphite reactors now unique to the Russian Federation (apart from a single larger unit in Lithuania). The four oldest of these were commissioned in the 1970s at Kursk and St. Petersburg and are of some concern to the western world. A further unit is under construction at Kursk;
- four small graphite-moderated BWR reactors in eastern Siberia, constructed in the 1970s for cogeneration (EGP-6 models);
- one BN-600 fast-breeder reactor.

Apart from Bilibino, several other Russian reactors also supply district heating — a total of over 8 PJ/a in addition to electricity.

Generally, reactors are licensed in the Russian Federation for 30 years after reaching first full power. Late in 2000, plans were announced for licensing extensions of twelve first generation reactors (Leningrad 1&2, Kursk 1&2, Kola 1&2, Bilibino 1-4, Novovoronezh 3&4) totalling 5.7 GW(e), and the extension period envisaged is now 15 years, necessitating major investment in refurbishing them starting 2006. So far, three 15-year extensions have been achieved for Novovoronezh 3&4, Kursk 1&2, Kola 1&2 and Leningrad 1&2. The Bilibino 1&2 units have been granted a 5-year licence extension. Replacement of all these twelve units after 2015–2020 is planned.

In 2006, Rosatom announced it was considering licensing extensions for and uprating of its eleven operating RBMK reactors. Following significant design modifications made after the Chernobyl accident, as well as extensive refurbishment including replacement of fuel channels, a 45-year lifetime is seen as realistic for the 1000 MW(e) units. In 2005, the RBMKs provided 48% of the Russian Federation's nuclear-generated electricity. The R&D Institute of Power Engineering is preparing plans for a 5% uprating of some of the units — at Leningrad, Kursk and Smolensk. Several more reactors have been under construction. A

mid-2006 announcement pledged US $665 million in 2007 towards completing Rostov/Volgodonsk-2, Kalinin-4 and Beloyarsk-4.

In September 2006, Rosatom announced a target for nuclear power providing 23% of electricity by 2020, thus commissioning two 1200 MW(e) plants per year from 2011 to 2014 and then three per year until 2020 — some 31 GW(e) and achieving some 44000 MW(e) of nuclear capacity then.

In October 2006, the Russian Federation formally adopted a US $55 billion nuclear energy development programme, with US $26 billion of this to 2015 coming from the federal budget. The balance will be from industry (Rosatom) funds and no private investment is involved. The Minister of Finance strongly supported the programme to increase nuclear share from 15.6% to 18.6% of total, hence improving energy security as well as promoting exports of nuclear power technology. After 2015, the complete funding of the nuclear power programme will come from Rosatom revenues.

Apart from completing units under construction there will be three standard third generation WWER reactors built: at Leningrad (two units as stage 2) and Novovoronezh (unit 6) to be commissioned 2012–2013. The first unit at each site is expected to cost US $3.0–3.7 billion.

In April 2007, the government approved in principle a construction programme to 2020 for electricity-generating plants. It is designed to maximise the share of electricity produced by nuclear, coal, and hydro while reducing the role of natural gas. Putting the planned units together with Kursk 5 unit under construction one gets:

TABLE 2.8. MAJOR POWER REACTORS UNDER CONSTRUCTION AND PLANNED IN THE RUSSIAN FEDERATION

Plant	Type	MW(e)	Status	Commercial operation
Volgodonsk 2	V-320	1000	Construction	2009
Kursk 5	RBMK-1000	1000	Construction	2010
Kalinin 4	V-320	1000	Construction	2011
Beloyarsk 4	BN-800 FBR	800	Construction	2012
Novovoronezh II-1	AES-2006/ WWER-1200	1200	Construction	2012
Novovoronezh II-2	AES-2006/ WWER-1200	1200	Planned	2013
Leningrad II-1	AES-2006/ WWER-1200	1200	Planned	2013
Volgodonsk 3	AES-2006/ WWER-1200	1000	Planned	2014
Leningrad II-2	AES-2006/ WWER-1200	1200	Planned	2014
Kursk II-1	AES-2006/ WWER-1200	1200	Planned	2015
Leningrad II-3	AES-2006/ WWER-1200	1200	Planned	2015
Volgodonsk 4	AES-2006/ WWER-1200	1000	Planned	2016
Leningrad II-4	AES-2006/ WWER-1200	1200	Planned	2016
Kursk II-2	AES-2006/ WWER-1200	1200	Planned	2017
Smolensk II-1	AES-2006/ WWER-1200	1200	Proposed	2017
Kursk II-3	AES-2006/ WWER-1200	1200	Proposed	2018
Smolensk II-2	AES-2006/ WWER-1200	1200	Proposed	2018

TABLE 2.8. MAJOR POWER REACTORS UNDER CONSTRUCTION AND PLANNED IN THE RUSSIAN FEDERATION (cont.)

Plant	Type	MW(e)	Status	Commercial operation
Kursk II-4	AES-2006/ WWER-1200	1200	Proposed	2019
Smolensk II-3	AES-2006/ WWER-1200	1200	Proposed	2019
Kola II-1	AES-2006/ WWER-1200	1000	Proposed	2020
Smolensk II-4	AES-2006/ WWER-1200	1200	Proposed	2020

2.2.9. Current status of nuclear power in the USA

Although no new plants are constructed in the USA, uprates of existing power plants continued to increase installed capacity (as of 2005, 4.2 GW(e) of net capacity was added, application for additional 1.1 GW(e) are pending). As of 2007, the USNRC granted 20-year licence extensions for 48 reactors and is working on additional licence applications for 27 reactors. The nuclear power plant Browns Ferry-1 (shut down since 1985) was restarted in 2007. Recently, several consortia of utilities have applied for construction and operation licences of new reactors (Generation III+).

The nuclear policy of the US government has recently changed as demonstrated by the GNEP initiative (see Section 2.6.2).

2.2.10. Current status of other nuclear power programmes

In this section a short summary of other ongoing nuclear power programmes in the world (not presented before) is provided (based on Ref. [9]).

Argentina

Decision to restart construction of Atucha-2 has been made by the government (plant is ~80% complete).

Belgium

In 2003, the government's policy to phase out nuclear power was confirmed by the Senate. The first reactor shut down would occur around 2015. Despite the phase out policy upgrading of the 7 operating units is ongoing.

In 2007, for the government a report has been issued by a commission for the analysis of the Belgian energy policy towards 2030 (Commission Energy 2030). The report emphasizes the disadvantages of the current nuclear phase out policy and recommends considering all energy options including nuclear especially to be able to fulfil the obligations of the Kyoto Protocol.

Brazil

Construction of Anga-3 is being planned.

Bulgaria

In November 2006, the Russian Federation and Bulgaria have signed an agreement on the construction of the Belene NPP. The construction project of Belene NPP implies commissioning of two WWER-1000 units.

Main advantages of WWER-1000 design for Belene NPP:

- The safety measures are a combination of traditional active and modern passive safety systems and includes also means for beyond design basis accident management.
- Optimal costs and length of construction time (commissioning of Unit 1 and 2 will be in 2011 and 2013 respectively).
- Balanced combination of Russian and European experience in nuclear engineering.

The design ensures obtaining higher performance characteristics by means of:

- reduction of outage time;
- extension of main equipment service life up to 60 years;
- increase of installed power factor up to 90%;
- capabilities of heat generation for home heating demand.

Czech Republic

The nuclear unit Temelin-2 (WWER-1000) was added to the grid in 2003.

Finland

Okiluoto was selected as the site for the 5th nuclear power plant in Finland. The construction of the new unit — a 1600 MW(e) EPR — started in 2005.

Germany

The phase out policy is still being maintained. As a result of this policy, two units (Stade and Obrigheim) have been shut down in 2003 and 2005, respectively.

Hungary

Plans have been announced to uprate by about 10% and to extend the lifetime of the Paks nuclear units for up to 20 years.

Islamic Republic of Iran

Construction of Busher-1 is finished. The construction of Busher-2 is planned.

Kazakhstan

A nuclear reactor is planned to be built with operation expected to start in 2015.

Lithuania

One of the two nuclear units (RBMK) was shut down in 2004 in accordance with an agreement with the European Union. The second unit is planned to be shut down in 2009. The government is in the planning stage for a new reactor.

Mexico

Uprating of the existing two BWR units is planned.

Netherlands

The planned shutdown of the single nuclear unit in the country was changed and the plant got a licence to operate at least till 2013. It is further planned to extend the operating licence till 2033.

Pakistan

A third nuclear reactor is to be built with operation expected to begin in 2010. A licence extension of 15 years has been authorized for the nuclear unit Kanupp.

Romania

Start of operation of Cernavoda-2 (CANDU) plant is planned in 2007. The government plans additionally to construct Cernavoda-3 (CANDU).

Slovakia

One plant (Bohunice-1) has been shut down in 2006 and a second one (Bohunice-2) is planned to be shut down in 2008 as part of an agreement with the European Union. Plans have been announced to complete nuclear units Mohovce-3 and 4 with operation expected to begin after 2009.

South Africa

Development of the pebble bed modular reactor (PBMR) is ongoing (see Section 2.4.6). Additionally, plans are being developed to add additional LWRs to the existing two PWR units.

Spain

The government is planning to enact a phase out policy but no specific time table has been defined. One unit (Jose Cabrera) was shut down in 2006.

Sweden

The government remains committed to the phase out policy over the next 30 to 40 years. However, power uprates of the existing units are planned.

Switzerland

In 2005, a new nuclear law was adopted that keeps open the option for deploying new nuclear units and does not limit lifetime of existing units.

Ukraine

One unit (Khmelnitsky-3, WWER-1000) is under construction (expected start of operation in 2012) and a second one is planned to be completed shortly. A lifetime extension of 15 years of the operating nuclear units is planned. Additional 20 TW(e) nuclear capacity should be commissioned until 2030.

United Kingdom

Due to economical reasons (low electricity prices) nine gas cooled power reactors (Magnox type) were shut down till 2007. The remaining gas cooled reactors (AGR type) are scheduled to be shut down beginning in 2011 till 2023. The PWR at Sizewell is supposed to be shut down in 2034. The option to build new nuclear power plants in the future is being discussed in public hearings. In 2007, the UK government announced to include several new nuclear units (replacing old ones) into their energy plan to assure a balanced supply of energy and reduce greenhouse gas emissions.

2.3. Development of water cooled reactors

This and the following sections provide some illustrative examples of reactor designs currently under development in different countries in the form of a short overview. Detailed information on the status of ongoing development of nuclear reactors can be found in Refs [10–14].

At first, the water cooled designs will be laid out. During the last decades continuous technical development of the classical water cooled nuclear reactor systems has been

performed. In the following sections examples of such advanced LWR as well as HWR systems are shortly presented.

2.3.1. WCR developments in Argentina

Since more than a decade an innovative small PWR concept is under development in Argentina, which is called CAREM (Fig. 2.5). The main design goal is to create a small reactor system suitable for developing countries with small grids. Also a special focus is on the inherent safety of the system.

Main features
- Thermal power: 100 MW(th)
- Electric power: 25 MW(e)
- Simple, modular and safe design
- Integrated primary circuit
- Natural convection cooling
- Self-pressurization
- Passive shutdown and emergency cooling systems
- Fuel: Enriched uranium
- Moderator/coolant: Light water

Applications
- Electric power generation
- Water desalinization

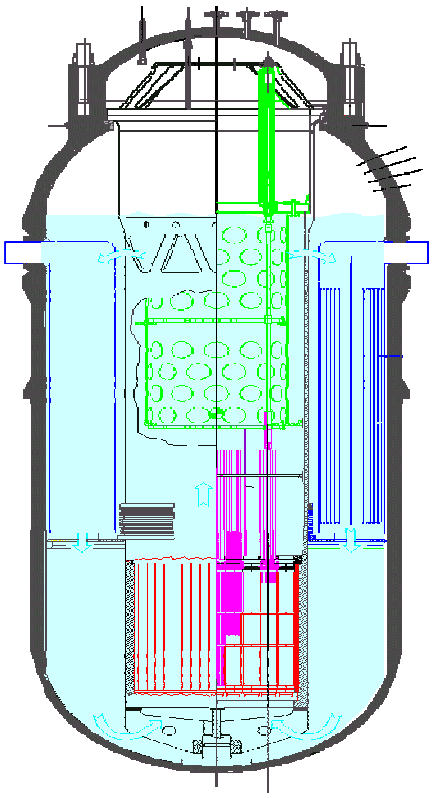


FIG. 2.5. CAREM PWR concept [15, 16].

The CAREM primary system is fully contained inside the pressure vessel, to reduce the possibility of a large loss of coolant accident. Also the reactor concept foresees:

- a self-pressurized system with passive safety (natural circulation);
- internal control rod drives;
- other safety features, like two independent shutdown systems (AgInCd and boron, both gravidity driven), a residual heat removal system, an emergency injection system and a pressure suppression type containment.

The CAREM-25 type will operate at 100 MW(th) (27 MW(e)), with low enriched uranium fuel (initial enrichment 3.4%) [16]. Recently, studies have been performed to increase the power output up to 300 MW(e).

2.3.2. WCR developments in Canada

Canada has continued to build on the success of its proven CANDU-6 design (Fig. 2.6), a pressure tube reactor with heavy water as moderator and coolant. New CANDU-6 reactors, such as those built in China, include evolutionary improvements to the design and materials used in the reactor systems. Because of its pressure-tube design, high neutron economy and

on-line refuelling, it would be easily adaptable to a number of fuel cycles ranging from the use of natural uranium, slightly enriched uranium, recovered uranium from reprocessed PWR or BWR fuel, MOX fuel, or direct use of spent PWR fuel (the DUPIC fuel cycle) in CANDU reactors. The reactor would also be capable of utilizing thorium-based fuel, or even inert matrix fuels. The latest design of this type of CANDU reactor is called CANDU-9 with a capacity of about 900 MW(e).

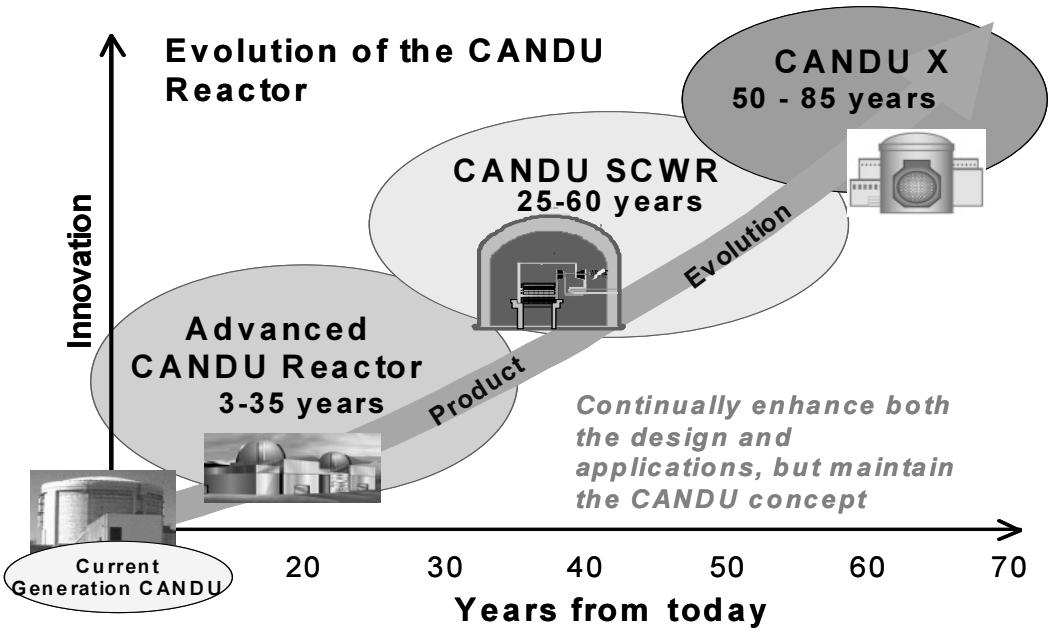


FIG. 2.6. Evolution of the CANDU reactor (starting in 2004).

Advanced CANDU reactor (ACR-700)

The ACR (Fig. 2.7) is a heavy water moderated but light water cooled, pressure-tube reactor with a smaller lattice pitch than the CANDU-6 reactor. Like earlier CANDU reactors, the ACR has a small, simple fuel bundle design and on-power refuelling. It shares the fuel cycle flexibility of the CANDU-6 and, because of its larger size, has somewhat improved neutron economy and achieves higher fuel burnup. The ACR also offers a much lower capital cost than present reactors and a lattice design that can provide a coolant void reactivity. As well, ACR fuel cycles will exploit the enhanced thermal-hydraulic properties of the advanced CANFLEX fuel bundle. The reactor is designed to run on low enriched uranium with high burnup, but is also exceptionally well-suited for burning MOX fuels. The original design of 700 MW(e) capacity has been extended to 1200 MW(e).

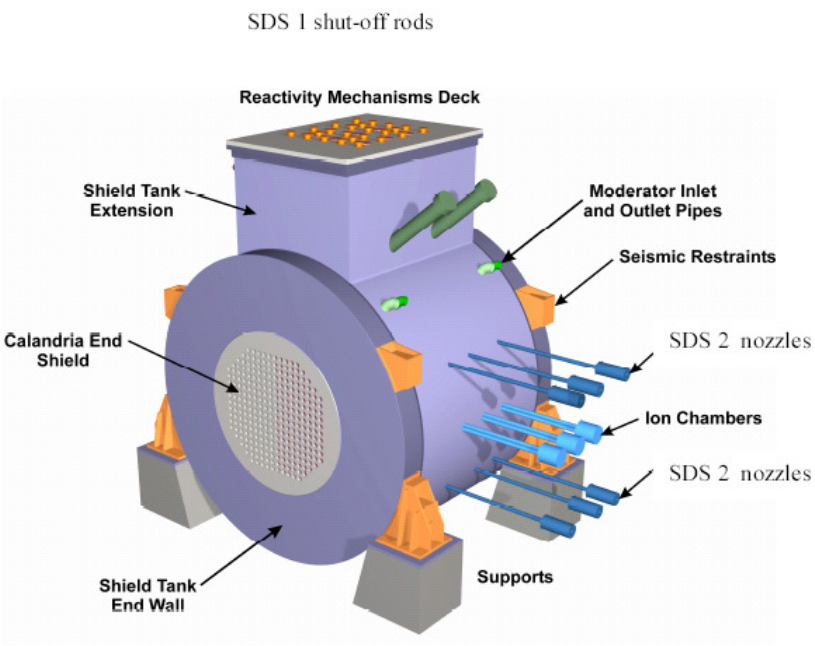


FIG. 2.7. ACR-700 reactor assembly [17].

Supercritical water reactor (SCWR)

The fuel channel SCWR (Fig. 2.8) is being developed within the Generation IV framework (to be discussed further in Section 2.6.1). This reactor extends the design concepts of the ACR to supercritical water conditions, to achieve higher thermodynamic efficiency and reduced capital cost.

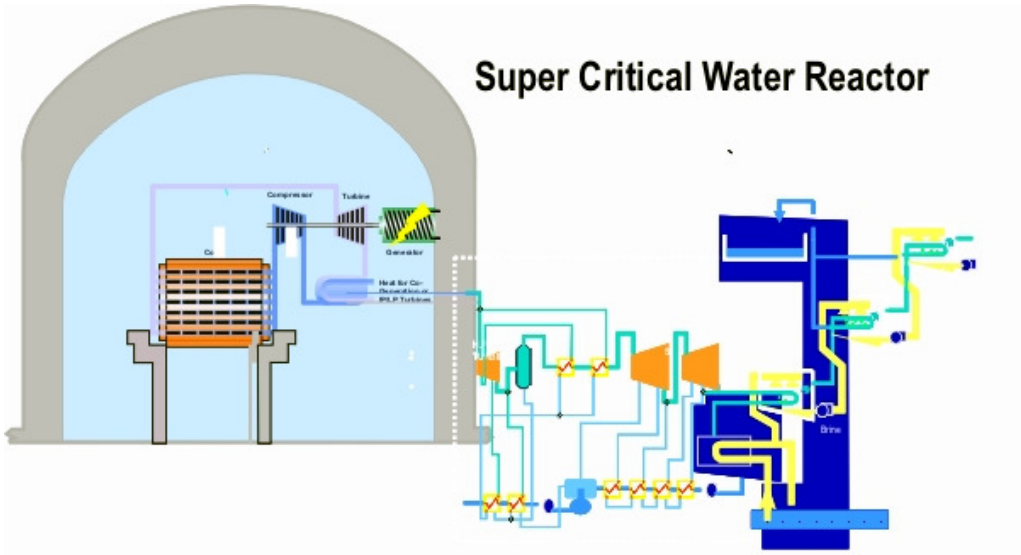


FIG. 2.8. Canadian supercritical water reactor.

2.3.3. WCR development in China

In China, a 1000 MW(e) PWR is being developed called CNP1000. It is based on a standardized 3 loop PWR. Some modification have been incorporated into this design, e.g. to decrease cost and increase the safety level with the goal to fulfil some internationally recognized utility requirements as for example the EPRI URD and the EUR.

A medium sized PWR called AC-600 is been developed using the design of the nuclear units Qinshan as a reference. The goal is to simplify the plant design to achieve economic competitiveness but in parallel increase the safety level. A similar PWR design called QS-600/W is under investigation, also using the Qinshan units as a reference, but with the additional capability to generate electricity and potable water. Also a 200 MW(th) integral PWR reactor for heat and desalination — called NHR-200 — has been designed and engineering studies concluded in mid-2006.

2.3.4. European WCR developments

Three water cooled reactor designs have been developed that meet the European Utility Requirements (EUR) of French and German utilities, with very stringent safety requirements: EPR, ABWR and SWR-1000.

EPR

The first one is the large European Pressurized Water Reactor (1550 to 1750 MW(e)), called EPR[4], because it is a follow-up development to a joint German–French development started already in 1995 under this name (Fig. 2.9). The goals of this new large EPR are very ambitious: it should generate 10% cheaper power, reach a fuel burnup of 65 GW·d/t·U, and achieve 36% thermal efficiency and 92% availability over a 60 years life time.

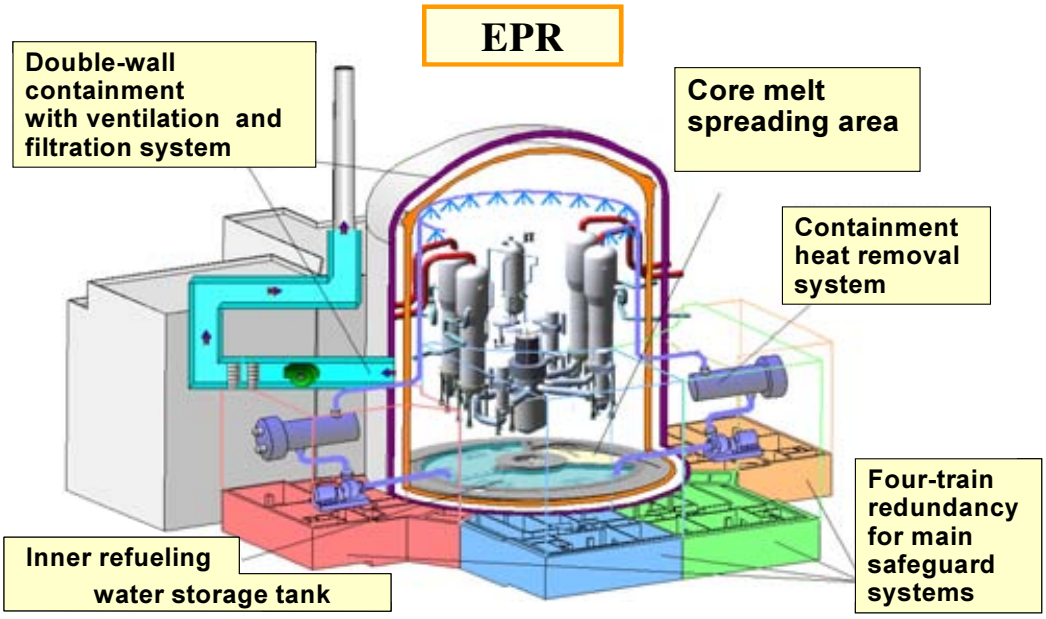


FIG. 2.9. Main features of the EPR.

SWR-1000

An innovative BWR has been developed by Siemens, called SWR-1000 with a power range of 1000 to 1300 MW(e). Many passive safety features have been added up to control accidents with a melting core. High burnup fuel can provide fuel cycle lengths of up to 24 months.

[4] In the meantime EPR is also interpreted as Evolutionary Pressurized Reactor.

ABWR 90+

The ABWR 90+ has been developed in Sweden by Westinghouse together with Scandinavian utilities. The size is planned to be 1500 MW(e) and it also shall meet the above-mentioned EUR requirements.

At CEA in France the concept of a medium sized integral PWR, called SCOR, with around 630 MW(e) is being developed [11].

2.3.5. WCR developments in India

The programme in India for advanced nuclear power systems has to be understood on the background of two specific targets for the future nuclear fuel cycles in India (Fig. 2.10):

- the use of the rich thorium resources in India;
- to minimize the long term burden of waste.

India is following a closed nuclear fuel cycle linking their three-stage [18] nuclear power programme based on PHWR, LMFBR and ^{232}Th – ^{233}U fuelled advanced self-sustaining reactor, for judicious utilization of indigenous modest uranium but vast thorium resources.

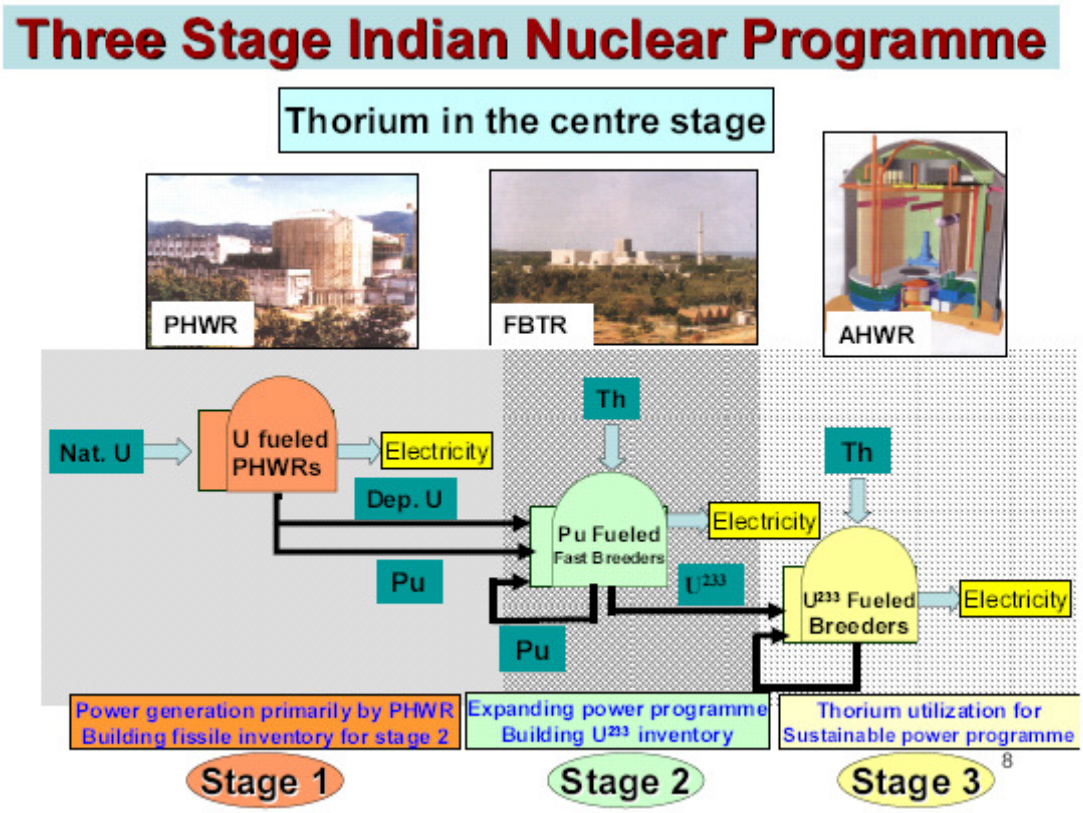


FIG. 2.10. Roadmap for a future Indian nuclear energy system (TWG NFCO 2007).

The PHWR fuelled with natural uranium oxide is the backbone of the programme. The plutonium and depleted uranium from these reactors will be utilized in liquid metal cooled fast breeder reactors (LMFBR) in the form of mixed uranium plutonium ceramic fuel. In the LMFBRs, initially depleted uranium but later thorium would be used as blanket material; the ^{233}U thus produced would be used in combination with ^{232}Th as self-sustaining fuel in some advanced thermal reactor system.

AHWR-300

As part of long term thorium utilization programme in the third-stage, detailed design of an advanced heavy water reactor of 300 MW(e) (AHWR 300) has been completed (Fig. 2.11). Like the Canadian ACR the 250 MW(e) design is light water cooled. The main part of the core is subcritical with Th/^{233}U oxide, mixed so that the system is self-sustaining in ^{233}U. A few seed regions with conventional MOX fuel will drive the reaction and give a negative void coefficient overall. The AHWR will utilize a unique 54 pin fuel assembly consisting of thirty (Th, ^{233}U)O_2 and twenty-four (Th, Pu)O_2 fuel pins.

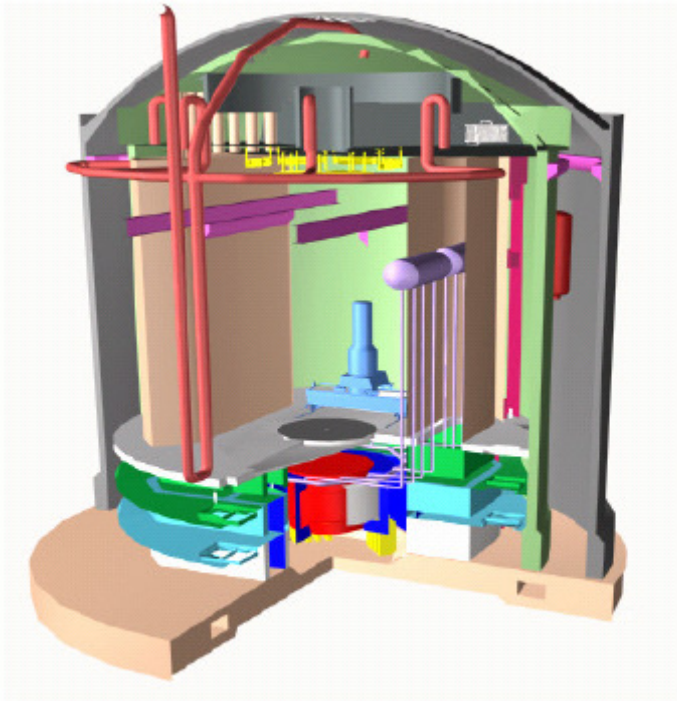


FIG. 2.11. Advanced heavy water reactor of 300 MW(e).

The AHWR includes various important innovative features for improved safety and economy, e.g.:

- natural circulation heat removal under normal operation and shutdown conditions;
- low core power density;
- slightly negative void coefficient of reactivity;
- direct spray of Emergency Core Cooling System (ECCS) water into fuel pins during Loss of Coolant Accident (LOCA);
- advanced accumulator with fluidic device for ECCS;
- gravity driven cooling system ensuring core cooling for 3 days without operator intervention during LOCA;
- passive containment cooling and isolation;
- utilization of moderator heat;
- utilization of low grade heat for desalination.

2.3.6. WCR developments in Japan

In Japan several types of water cooled reactors have been designed or are under development.

Advanced pressurized water reactor (APWR)

The 1500 MW(e) APWR design has been developed by four utilities with Mitsubishi (MHI) and (earlier) Westinghouse. The APWR is in the process of being licensed in Japan with the prospect to be the first 1538 MW(e) units being constructed at Tsuruga (units 3&4) in 2009. Approval by Fukui prefecture was given in March 2004. The design is simpler than present PWRs, combines active and passive cooling systems to greater effect, and achieves over 55 GW·d/t burnup. Design work continues and will be the basis for the next generation of Japanese PWRs.

The next version is called APWR+ is a 1750 MW(e) unit and is designed for a full-core MOX capability. MHI is planning to market its 1750 MW(e) APWR in the USA. MHI intends to lodge an application for US design certification in 2008. MHI also participated in developing the Westinghouse AP-1000 reactor, but now that Westinghouse has been sold to Toshiba, MHI will develop PWR technology together with AREVA.

Advanced boiling water reactor (ABWR)

The Japanese version of the ABWR was developed jointly by GE, Hitachi Ltd and Toshiba Corp. under the sponsorship of the Tokyo Electric Power Company (TEPCO). Hitachi is also developing 600, 900 and 1700 MW(e) versions of the ABWR. The ABWR-II has a capacity of 1717 MW(e).

In 1987, TEPCO announced its decision to proceed with a two-unit ABWR project at its Kashiwazaki-Kariwa (KK) nuclear power station, 220 kilometres north-west of Tokyo, as power station unit KK-6&7 began commercial operation in November 1996 and July 1997, respectively. Each is rated 1315 MW(e) (net). The results of the first ten reactor years of combined operating experience for TEPCO KK-6&7 are indicated below:

- The ABWRs are performing up to expectations.
- Unplanned shutdowns have been due to conventional problems and do not suggest there are any ABWR-specific problems.
- BWR technology is becoming safer and more economic.
- Compared to earlier BWR technology, ABWRs have lower occupational radiation exposure, increased availability, higher load factors and lower O&M costs.
- ABWRs would operate more efficiently under less severe operating constraints and with improved management strategies.

In mid-2005, the Nuclear Energy Policy Planning Division of the Agency for Natural Resources and Energy instigated a 2-year feasibility study on development of next generation LWRs. The new designs, based on ABWR and APWR, are to lead to a 20% reduction in construction and generation costs and a 20% reduction in spent fuel quantity, with improved safety, a 3-year construction time, and a 60-year life time. Projected sizes range from 800 to 1700 MW(e).

An ABWR-II (Fig. 2.12) is being developed which will be even more economical than existing ABWRs. In order to cut the cost of plant construction per kW, it is most effective to increase the power output (1700 MW(e) class). The new design concept permits increasing power output without reducing design margins and improves plant safety and reliability [19, 20].

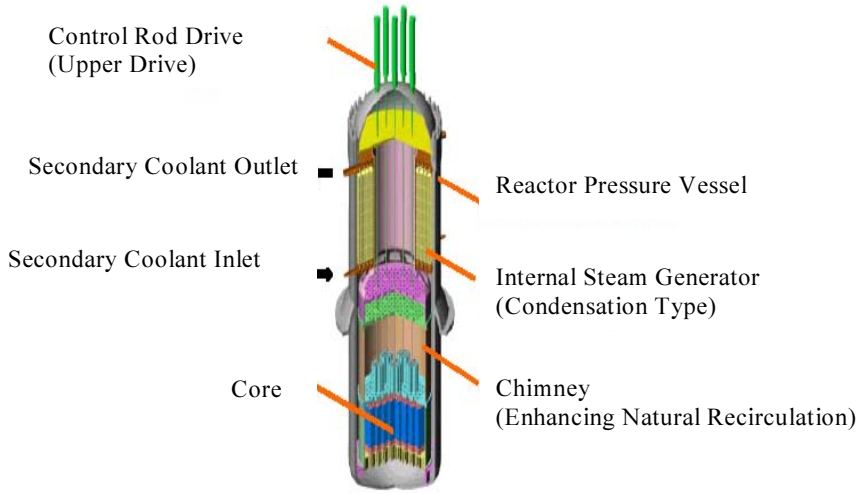


FIG. 2.12. ABWR II reactor pressure vessel and reactor internals.

Small and medium sized reactors (SMR)

In addition to the large reactors discussed above several concepts of water cooled SMRs are studied. Examples are a small BWR called CCR with 300 MW(e), an integral PWR with 350 MW(e), called IMR, and a pressure tube reactor called KAMADO with 300 MW(e) (for further details see Ref. [11]). The concept of a special type of SMR without on-site refuelling is also under investigation; examples are the PSRD and PFPWR50 concepts (for further details see Ref. [12]).

2.3.7. WCR developments in the Republic of Korea

In the Republic of Korea, two tracks have been considered for the enhancement of commercial nuclear systems on the basis of PWR technology. One is the development of advanced large size PWR, called APR-1400, and the other one is a small size PWR called SMART for dual uses of electricity generation and seawater desalination.

APWR-1400

The APR-1400 is a two-loop 1400 MW(e) pressurized water reactor, which is an extension of the current Korean Standard Nuclear Power Plant (KSNP) design (Fig. 2.13). Numerous changes from the KSNP design had been made in the component sizes and design parameters primarily due to the increase in power. The APR-1400 obtained a design certification from the Korean national regulatory agency in May 2002, and now is placed in construction preparation and scheduled to reach its first commercial operation by 2013.

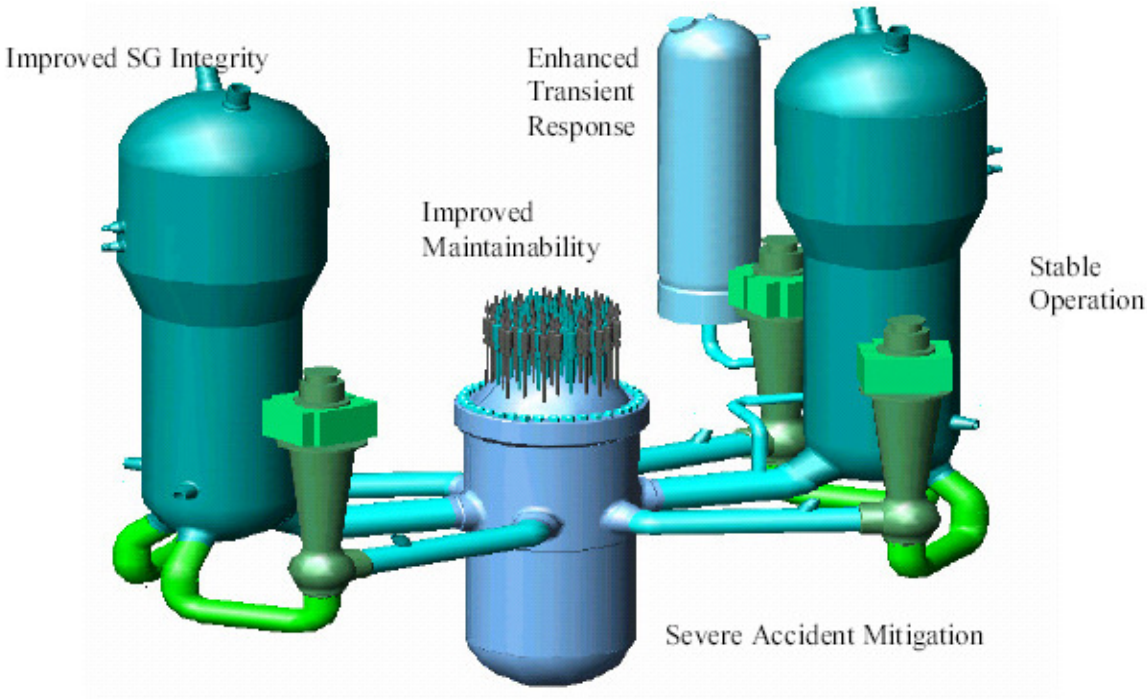


FIG. 2.13. APR-1400 design concept

SMART

The SMART (system-integrated modular advanced reactor), is an innovative technology-embedded PWR for seawater desalination and electricity generation (Fig. 2.14). In 2002, KAERI (Korea Atomic Energy Research Institute) successfully completed the basic design of 330 MW(th) SMART as a part of national R&D project. In 2003, KAERI and Doosan Heavy Industry launched a design & construction project of a 65 MW(th) pilot plant named SMART-P.

Recently, the Republic of Korea has announced that instead of constructing the pilot plant SMART-P they are considering to build a reference plant SMART with the full power of 350 MW(e).

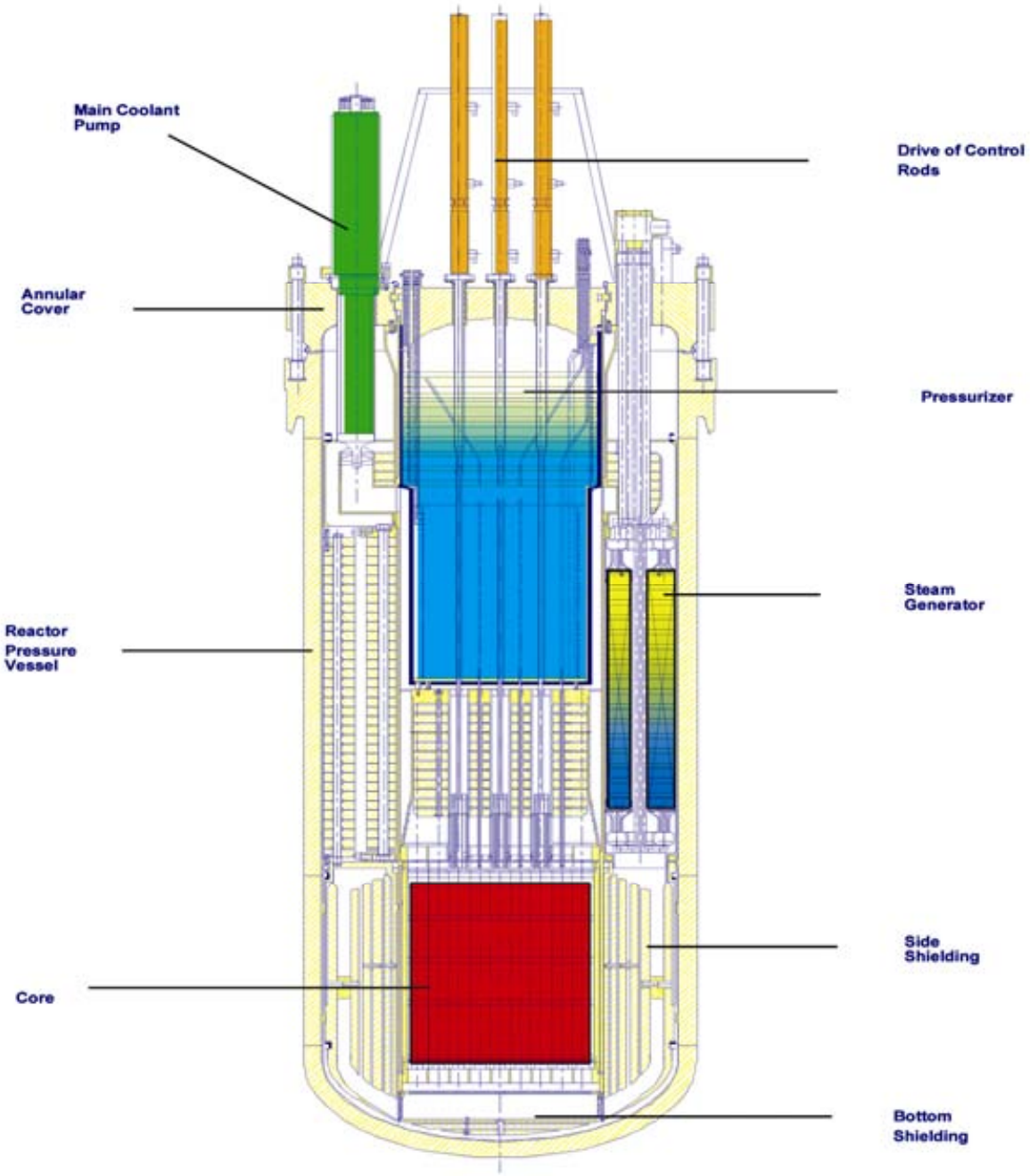


FIG. 2.14. SMART, system-integrated modular advanced reactor.

2.3.8. WCR developments in the Russian Federation

Several types of WCR were developed in the Russian Federation.

WWER-1000 V-392 reactor

An advanced WWER-1000 (V-392) with 1000 MW(e) (Fig. 2.15) was developed and is under construction at Novovoronezh.

The basic difference of this design from other designs of WWER with high power output consists of applying advanced equipment and implementing of additional passive safety systems in combination with active and traditional passive systems. The design improves reliability of equipment and allows preventing and mitigating consequences of design basis and beyond design basis accidents (BDBA) more effectively.

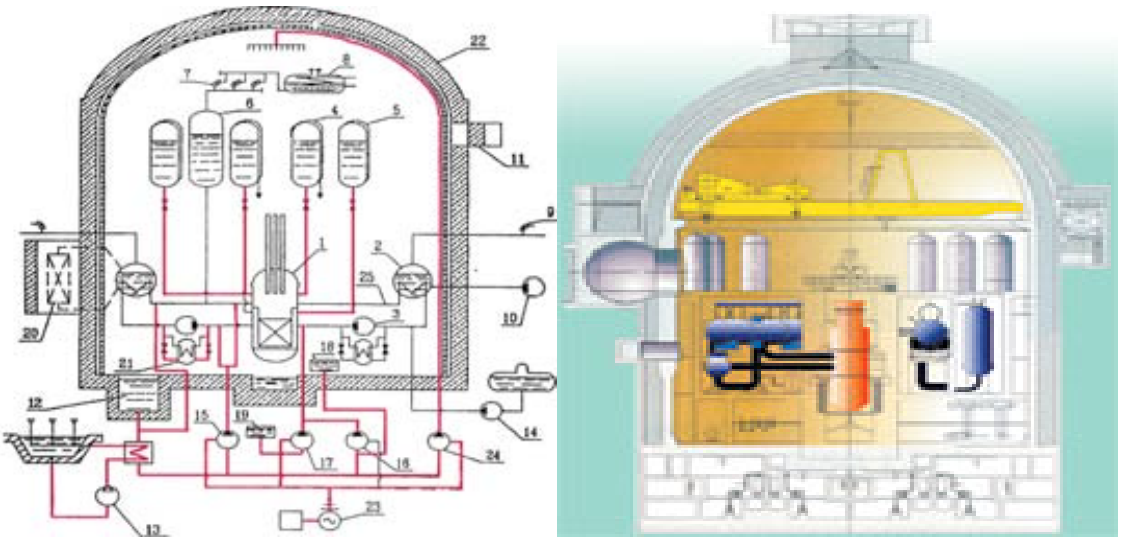


FIG. 2.15. Concept of WWER-1000 (V-392) with 1000 MW(e).

In addition another advanced WWER-1000 (V-91) has been developed with western control systems. Two of them are being built in China at Jiangsu Tianwan, one in the Islamic Republic of Iran, and two in India.

WWER-1500 (V-448) reactor

A WWER-1500 (Fig. 2.16) is being developed on the basis of the WWER-440 and WWER-1000 design. The conceptual design of the WWER-1500 adheres to the principles and approaches contained in the requirements of European Utility Requirements (EUR) and IAEA recommendations regarding the basic technical characteristics and safety level.

The design concept includes:

- application of WWER design experience with realization of the evolutionary succession principle;
- development of plant main equipment with orientation towards the Russian industrial base;
- consideration of work results on enhancement of equipment reliability and increase of service life in operating WWER-1000 and WWER designs of new generation;
- maximum usage of R&D achievements for WWER-1000;
- introduction of alternative solutions subsequently to their comprehensive and representative perfection;
- following the basic principles of safety, including requirements of domestic and foreign standards and regulations, i.e. IAEA recommendations, EUR standards, and application of certified calculation procedures and programmes;
- ensuring economic competitiveness.

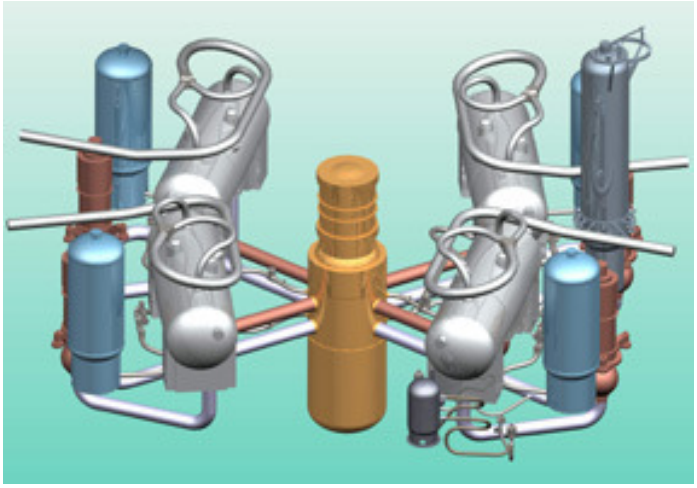


FIG. 2.16. Sketch of WWER-1500 reactor.

The WWER-1500 design of the steam generators (Fig. 2.17) allows the following provisions:

- enhancement of reliable operation of the tubing due to improvement of natural circulation;
- enhancement of the efficiency of sludge removing and washing-off at the vessel bottom and inter-tube space;
- enhancement of easiness of tubing test and examination from the secondary side;
- enhancement of reliability and convenience of tubing metal automatic control due to increase of bending radius.

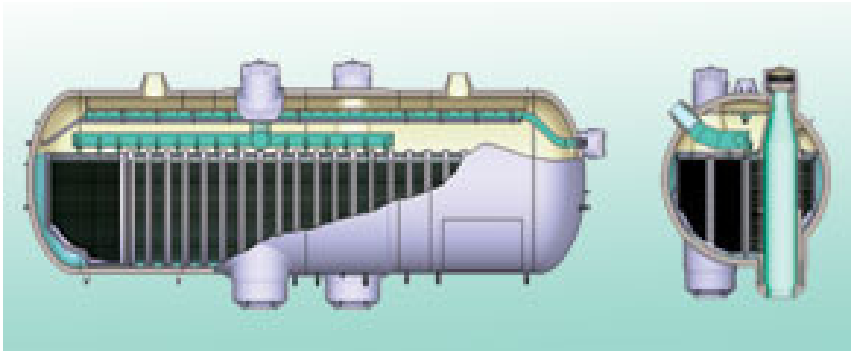
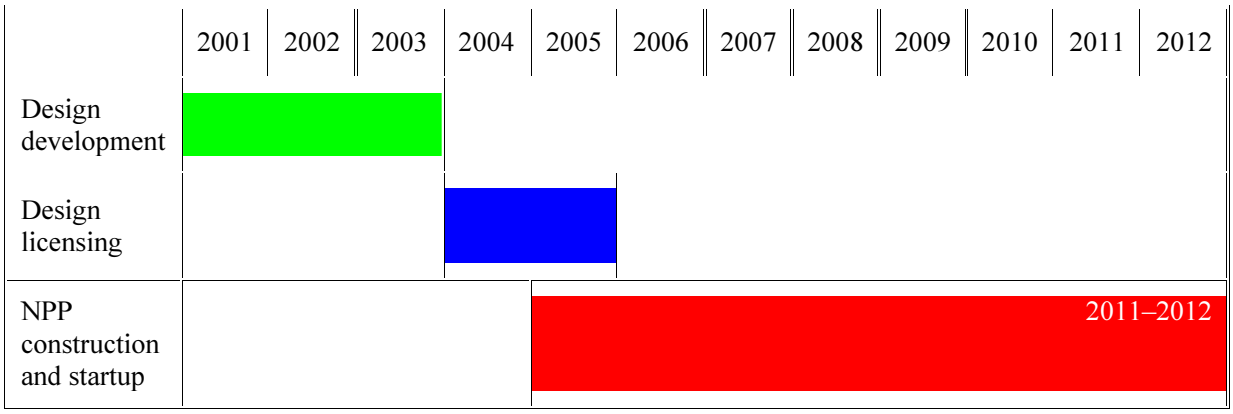


FIG. 2.17. Steam generator of WWER-1500.

Table 2.9 shows the schedule for the development and deployment of the WWER-1500 reactor.

TABLE 2.9. DEVELOPMENT SCHEDULE FOR AN NPP WITH A WWER-1500 REACTOR



KLT-40C reactor

The KLT-40C [21] takes advantage of the experience gained through the operation of the KLT-40 reactors used to provide power for icebreaker propulsion. The KLT-40C utilizes a PV/loop nuclear steam plant configuration similar to a conventional PWR, incorporating forced reactor coolant circulation at power (Table 2.10). The major unique aspect of the KLT-40C nuclear power plant design is the incorporation of two KLT-40C reactors into a comprehensive barge, referred to as the floating (Fig. 2.18) nuclear power system.

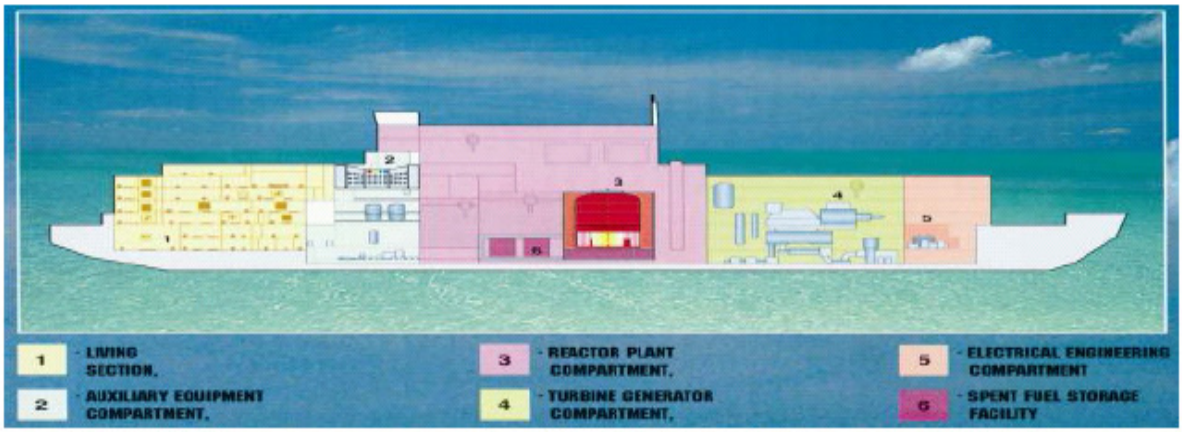


FIG. 2.18. Russian floating nuclear power system based on KLT-40C nuclear reactor.

TABLE 2.10. BASIC CHARACTERISTICS OF NSSS WITH KLT-40C REACTOR

Characteristics	Values
Number of reactors	2
Reactor type	Pressurised water
Reactor heat generating capacity, MW	2 × 135
Steam generating capacity, tonnes/hour	2 × 240
Steam temperature, °C	290
Feed water temperature, °C	170
Steam pressure, MPa	3.7
Containment type	Single, steel

The engineers of OKBM and JSC Malaya Energetica (Russian Federation) and CANDESAL Technologies (Canada) have joined efforts to develop a floating energy block KLT-40 reactor that will be coupled with a floating desalination system.

Rosatom is also planning to construct seven further floating nuclear power plants in addition to the one now under construction, each with two 35 MW(e) KLT-40S nuclear reactors. Five of these will be used by Gazprom for offshore oil and gas field development and for operations on the Kola and Yamal peninsulas. One is planned for 2012 commissioning at Pevek on the Chukotka peninsula, another for Kamchatka region, both in the far east of the country. Further far east sites being considered are Yakutia and Taima. Electricity cost is expected to be much lower than from present alternatives.

VBER-300

OKBM's VBER-300 PWR is a 295 MW(e) unit developed from naval power plants and was originally envisaged in pairs as a floating nuclear power plant. As a cogeneration plant it is rated at 200 MW(e) and 1900 GJ/hr for heat or desalination. The reactor is designed for 60 year life and 90% capacity factor. It is now planned to develop it as a land-based unit with Kazatomprom, with a view to exports, and the first unit could be built in Kazakhstan.

VK-300

The VK-300 boiling water reactor is being developed for both power (250 MW(e)) and desalination (150 MW(e) plus 1675 GJ/hr). It has evolved from the VK-50 BWR at Dimitrovgrad, but uses standard components wherever possible, e.g. the reactor vessel of the WWER-1000. A feasibility study on building four cogeneration VK-300 units at Archangelsk was favourable, delivering 250 MW(e) power and 31.5 TJ/a heat.

RUTA-70

This concept is being developed by IPPE and Rosatom jointly. It is a small integral PWR with 70 MW(th) to be used for district heating.

SMRs without on-site refuelling

Several concepts of water cooled small reactors with long intervals between refuelling are investigated in the Russian Federation, e.g. UNITHERM, ELENA, ABV, VKR-MT, etc. (for more details see Ref. [12]).

2.3.9. WCR developments in the USA

Advanced boiling reactor (ABWR)

As mentioned before, GE together with Toshiba and Hitachi developed the ABWR, two examples of which are now in commercial operation in Japan since 1996 and 1997, and two more started up in 2005, and two are under construction in Taiwan (China). This ABWR has also been certified meeting EPRI and European utility requirements for advanced reactors and its design was licensed also in the USA in 1997.

System 80+

The System 80+ developed by Westinghouse (formerly Combustion Engineering), an advanced pressurized water reactor, is now ready for commercialization. The Korean Standard Reactors incorporated some design features of the System 80+.

Advanced pressurized reactor AP-600

The AP-600 from Westinghouse is a small advanced PWR with innovative passive safety features, which decrease the probability of a core melt down accident by a factor of 100

compared to the present USNRC requirements. NRC granted final design certification for this reactor design in 1999. Simplification has reduced plant components and should reduce construction costs. The AP-600 has been bid overseas but has never been built. Westinghouse has de-emphasized the AP-600 in favour of the larger AP-1000, which has a potential to be even less expensive (on a cost per kilowatt or capacity basis).

Advanced pressurized reactor AP-1000

The AP-1000 (Fig. 2.19) is an enlargement of the AP-600, designed to almost double the reactor's target electricity output without proportionately increasing the total cost of building the reactor. Westinghouse anticipates that operating costs should be below the average of reactors now operating in the United States of America. While Westinghouse owns rights to several other designs, the AP-1000 is the principal product that the company now promotes in the United States of America for near term deployment. The AP-1000 includes innovative, passive safety features and a much simplified design intended to reduce the reactor's material and construction costs while improving operational safety. The AP-1000 design is favoured for construction at several potential sites in the United States of America. During 2007 or 2008, it is anticipated that the AP-1000 will be the subject of combined licence (COL) applications to build and operate new reactors in the United States of America. In early 2005, Westinghouse submitted a bid to build as many as four AP-1000s at two sites in China. In January 2006, the NRC approved the final design certification for Westinghouse Electric Co.'s AP-1000 advanced design reactor. The Chinese government announced in 2007 that two units are considered to be built at Haiyang in the Shandong province. The AP-1000 is also called EP-1000 (European passive plant) in Europe.

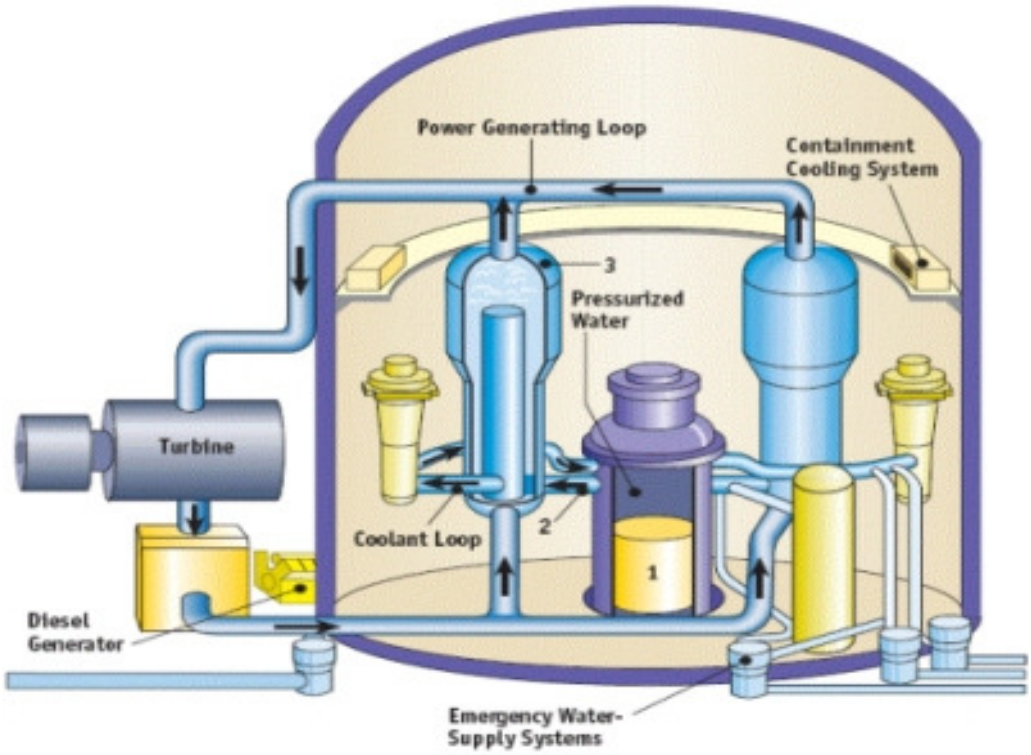


FIG. 2.19. AP-1000 design.

ESBWR

This is an advanced design (Fig. 2.20) developed by GE with a 1390 MW(e) output based on an earlier concept called SBWR. It incorporates innovative, yet proven features to further simplify (11 systems are eliminated from previous designs, 25% fewer pumps, valves and

motors) the inherently simple direct cycle nuclear plant. As an example of these new features, there is no need for reactor coolant pumps as the core flow is maintained during all operational stages via natural convection. Another example is the passive cooling system used for containment cooling after hypothetical accidents.

The design is currently in the US design certification process and submissions of construction and operating licence are expected in 2007.

SMR

Several concepts of SMRs without on-site refuelling are developed in the USA, e.g. MASL-WR and AFPR (for details see Ref. [12]).

Several additional designs are evaluated in the USA, as for example: A pebble bed boiling water reactor with superheated steam, using ceramic cladded microfuel elements (MFE), as proposed by Batelle, Pacific Northwest National Laboratory (PNNL) [22].

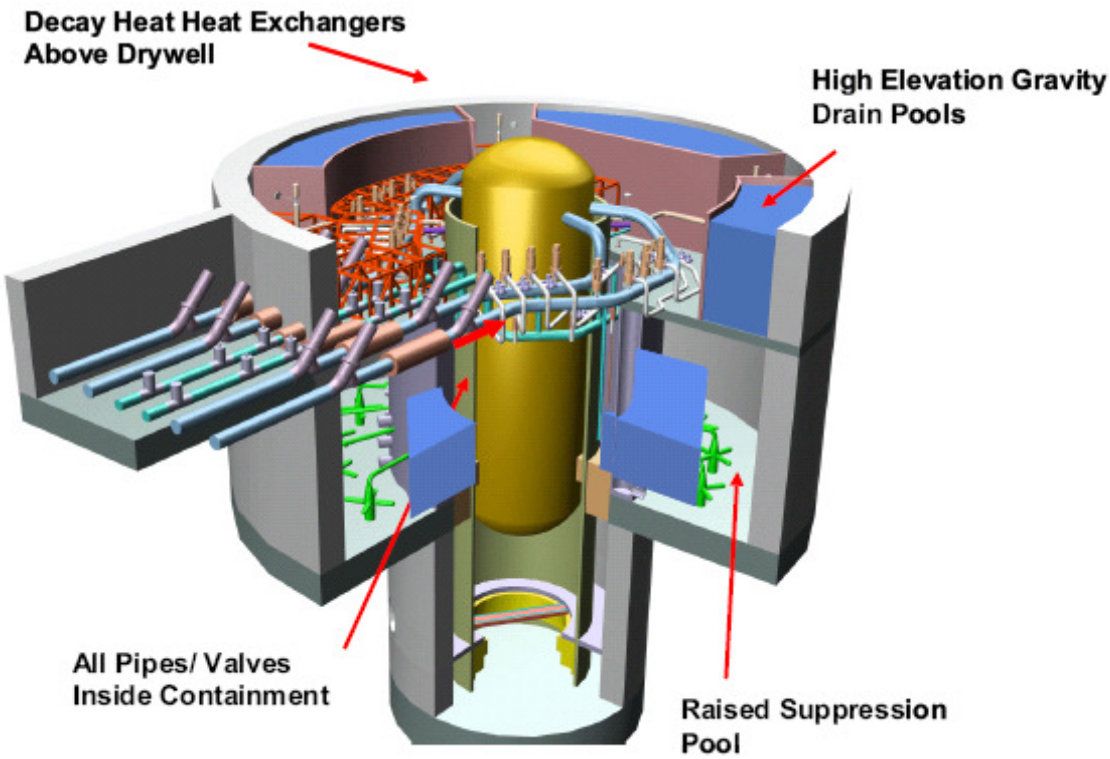


FIG. 2.20. ESBWR design features [10].

2.3.10. The international cooperative development programme IRIS

IRIS (International Reactor Innovative and Secure) is an advanced (pressurized) light water cooled modular reactor being developed by an international consortium consisting of 22 organizations, led by Westinghouse. It is a modular, integral, light water cooled, medium power reactor (335 MW(e)/module), (Fig. 2.21) with three innovative design features [23]:

- safety by design (e.g. no need of a spray system);
- optimized maintenance (48 months);
- long core life (4 years refuelling time).

IRIS provides a viable bridge to Generation IV reactors (to be discussed in Section 2.6.1) and has excellent capability to satisfy in the near/mid-term timeframe the Global Nuclear Energy Partnership (GNEP, to be discussed in Section 2.6.2) requirements for small scale reactors.

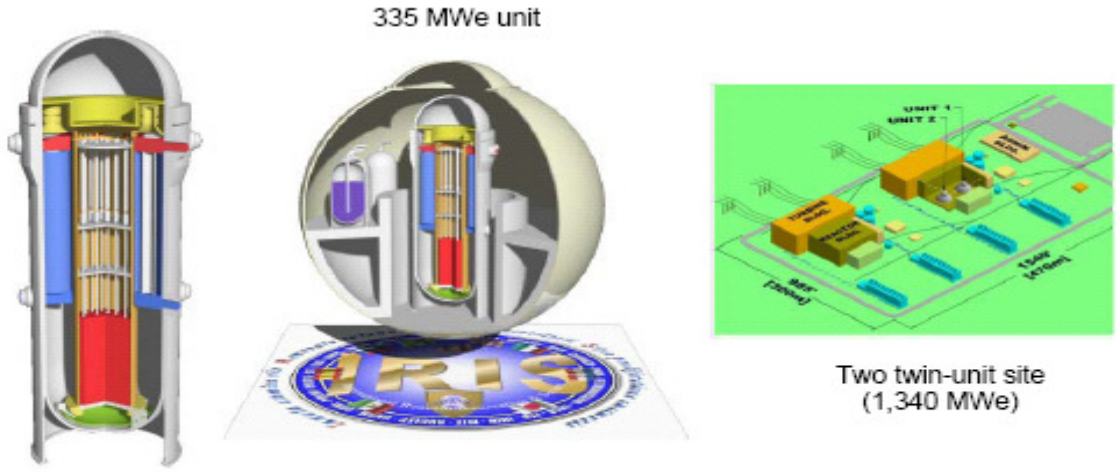


FIG. 2.21. The IRIS reactor design.

The 6.78 m outside diameter by 21.4 m in height IRIS integral vessel houses the reactor core, its support structures, upper internals, eight steam generators, internal shields, pressurizer and heaters, and eight reactor coolant pumps.

Because of the rather small size and the high level of inherent safety it is particular intended to be proposed for small and still developing countries.

2.4. Development of high temperature gas cooled reactors (HTGR)

In the following sections, examples of gas cooled reactor designs are shortly described that are in different stages of development. Most of design work of HTGR started already a long time ago but did not reach (full) commercial application up till now.

2.4.1. Short history of HTGR development

The development of HTGR began in the 1950s. The 20 MW(th) Dragon reactor operated in the UK 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, the 15 MW(e) AVR and the 40 MW(e) Peach Bottom HTGR were constructed and successfully operated in Germany from 1967 to 1988 and USA from 1967 to 1974, respectively. The AVR used uranium in coated particles in the form of balls as the fuel, the HTGR had a core with high enriched uranium mixed with thorium.

The commercialization was approached thereafter via the FSV-HTGR of 330 MW(e) in the USA (operated from 1976 to 1989) and the THTR of 300 MW(e) in Germany (operated from 1971 to 1989). Both reactors used a mixture of uranium and thorium as fuel.

Further continued interest in development of larger steam cycle HTGR plants included the German HTR-500, the Russian VG-400 and the US HTGR-SC, etc. However, these plants had not been constructed, mainly due to less economic competitiveness compared to LWRs.

In the 1980s, the 80 MW(e) HTR module (HTR-MODUL) concept was first developed by Siemens/Interatom, Germany, which had passive safety features, coupled with attractive characteristics of a modular concept.

The modular HTGR concept was followed by the gas turbine modular HTGR to improve the efficiency and economy, as exemplified by the South African PBMR Project and the USA/Russian GT-MHR Project (to be discussed later).

HTGRs potentially can use thorium-based fuel (Sections 3.4.2 and 3.5.1), such as highly enriched uranium (HEU) with Th, ^{233}U with Th, and Pu with Th.

From the beginning the purpose of the HTGR development was to generate helium temperatures as high as up to 950°C for direct industrial (e.g. chemical) application or to directly drive gas turbines to produce electricity with thermal efficiencies of up to 48%.

Currently, HTGR developments promise more practical designs; however, a key question is the integrity of the direct cycle system. Some of these national developments of HTGR underway are laid out in the following sections.

2.4.2. HTGR development in China

Research and development work in China on high temperature gas cooled reactors started in beginning of the 1970s. At that time, the research and development work was focused on gas cooled breeders using a thorium fuel cycle. The R&D and design work was carried out for a helium cooled thorium breeder of 100 MW output, using spherical fuel elements.

In March 1992, the state government approved the project of constructing the 10 MW pebble bed high temperature gas cooled test reactor (referred to as HTR-10). The high temperature gas cooled reactor — pebble bed module (HTR-PM) is a modular high temperature gas cooled reactor (HTGR) plant which is designed by the Institute of Nuclear and New Energy Technology (INET), Tsinghua University of China (Fig. 2.22). The current HTR-PM design falls into the category of innovative small and medium sized reactors, featuring 160 MW electrical output per module. The HTR-PM is being promoted for an industrial demonstration plant.

Basic design of HTR-10 was completed in 1994, and the construction permit was issued after licensing review of the Preliminary Safety Analysis Report and other relevant application documents. The construction of HTR-10 formally started with the first concrete being poured in June 1995. In parallel to the engineering design and construction work, a number of engineering tests were made, including tests for the fuel handling system, reactor shutdown system, once-through steam generator, etc. By the end of 1998, key components like pressure vessels, the steam generator and helium circulator were manufactured, and installation and commissioning work started. In the meantime, spherical fuel elements were manufactured and qualified through irradiation experiments. In December 2000, the HTR-10 reactor was loaded with nuclear fuel and went critical.

In January 2003, the HTR-10 reached full power of 10 MW. The HTR-10 reactor is currently in operation, and safety experiments are being performed. At the same time, development of helium turbine technologies is ongoing, as it is planned to couple a helium turbine power generation system to the HTR-10 reactor.

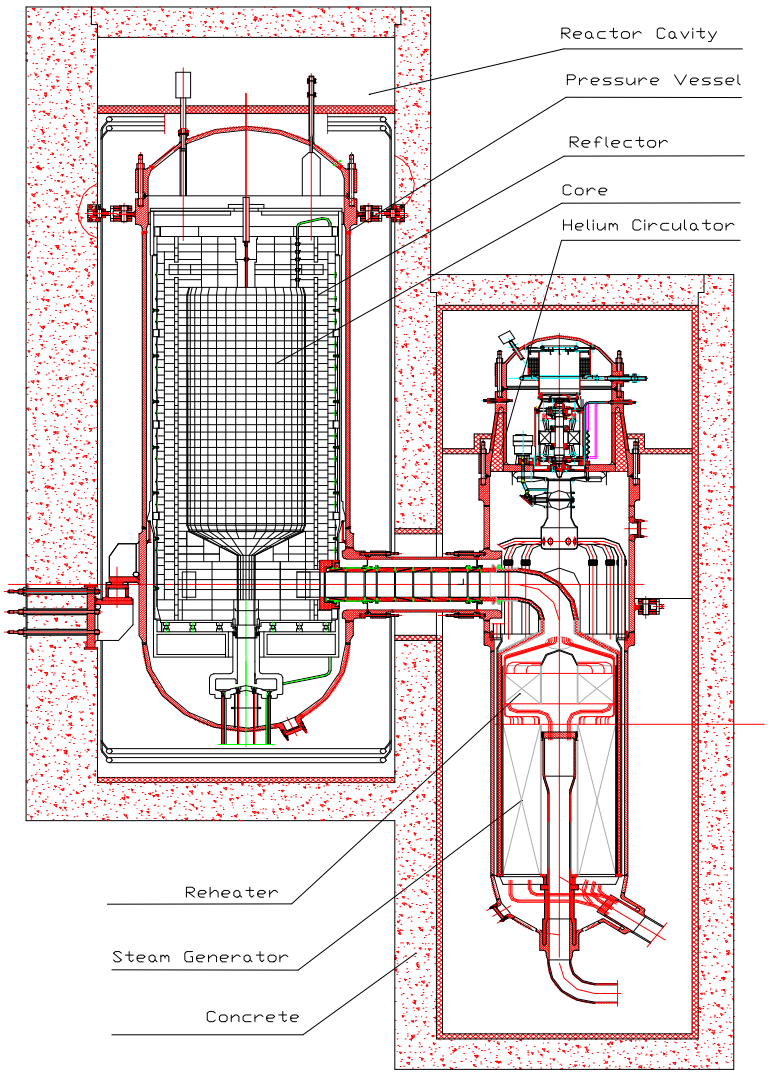


FIG. 2.22. High temperature gas cooled reactor – pebble bed module (HTR-PM).

With the HTR-10 being successfully constructed and operated, effort is being made to start developing an industrial modular HTGR demonstration plant. The commercial HTR units will be 160–200 MW(e) reactors with pebble bed fuel, similar to that being marketed by South Africa.

2.4.3. HTGR development in France

Besides continuing LWR and SFR development, presently the priorities with new advanced reactor technology focus on gas cooled systems (Fig. 2.23) which not only offer increased effectiveness for the production of electricity but also the option for process heat at very high temperatures of first 850°C (in the very high temperature gas cooled reactor, VHTR), and later of >950°C (in the gas cooled fast reactor, GFR, to be discussed further in Section 2.5.3). There are already quite detailed ideas on the necessary R&D, at least for the VHTR development (Fig. 2.24).

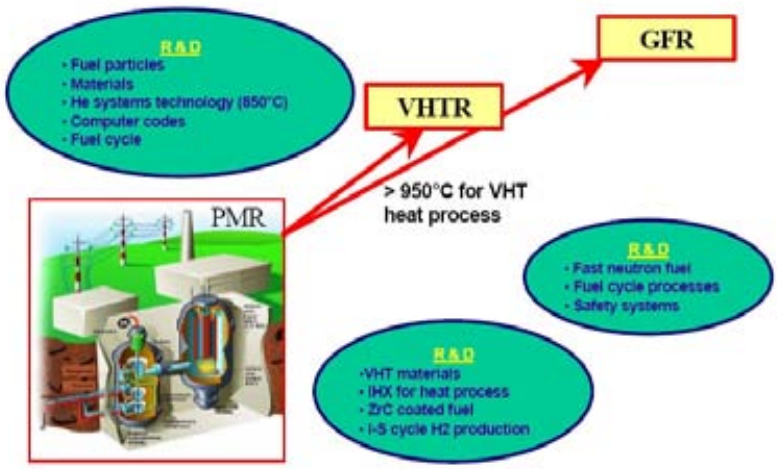


FIG. 2.23. Sequential development of gas cooled systems by CEA in France [24].

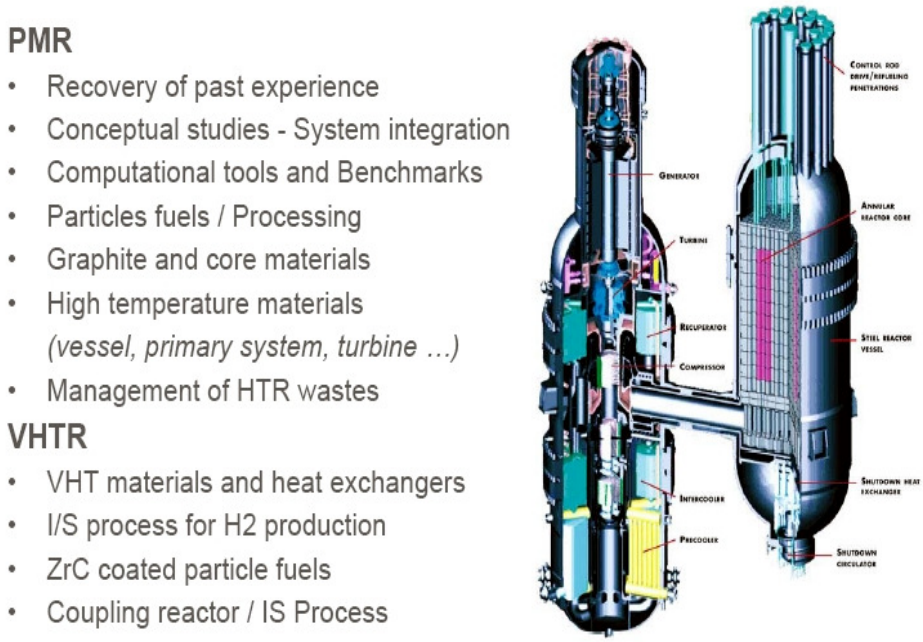


FIG. 2.24. French R&D for the systems PMR + VHTR.

A helium cooled reactor concept, called ANTARES [25], is developed by French industry; the design has a thermal power of about 600 MW(e), uses SiC TRISO UO_2 particle fuel, and is intended to provide a combination of power and process heat.

2.4.4. HTGR development in Japan

Interest in the HTGR as an advanced nuclear power source for cogeneration application of electricity production and high temperature heat for industrial process has resulted in the construction of the high temperature engineering test reactor (HTTR) at the Japan Atomic Energy Research Institute (formerly JAERI, now called JAEA) (Fig. 2.25). The HTTR attained the first criticality on 10 November 1998 and achieved its full power of 30 MW and a reactor coolant outlet temperature of about 800°C on 7 December 2001, and 950°C on 19 April 2004. The fuel uses a hexagonal block arrangement (prism). On the basis of the

HTTR, JAEA is developing a larger (600 MW(th) module) version directly coupled to a gas turbine for electricity production.

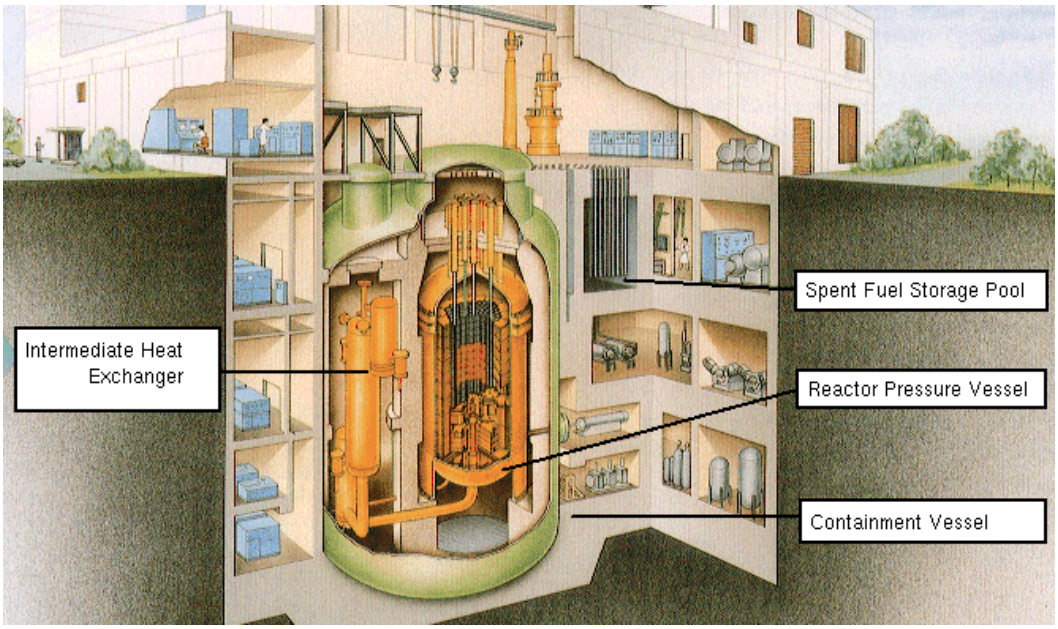


FIG. 2.25. Outlay of HTTR.

The purpose of the HTTR project is to establish and upgrade HTGR technology and enable irradiation tests for innovative basic research in the field of high temperature engineering. It is widely recognized in the nuclear community that the timely and successful operation of the HTTR and experiments using the HTTR are major milestones in HTGR development, and in the development of high temperature nuclear process heat application.

In addition to accumulation of the HTTR operation and performance data, extensive tests are planned in the HTTR, e.g. the tests for the development of advanced fuel and graphite materials as well as safety demonstration tests to verify the inherent safety of HTGR. A hydrogen production system is designed applying a steam reforming process of natural gas using nuclear heat (10 MW, 950°C) supplied by the HTTR [26] (Fig. 2.26). This system is tested in an out-of-reactor loop facility under construction to confirm the design of the heat application system and its performance prior to the HTTR coupling. In parallel to the HTTR project, some Japanese industries are exploring the feasibility of HTGR commercialization. A further industrial effort is expected to follow for commercialization HTGR.

SMR

Several modular gas cooled small reactor concepts are studied in Japan. One example is a 274 MW(e) unit, called GHTR-300, with prismatic fuel elements. Another is the FABIG-HTGR design with pebble bed fuel (for details see Ref. [11]).

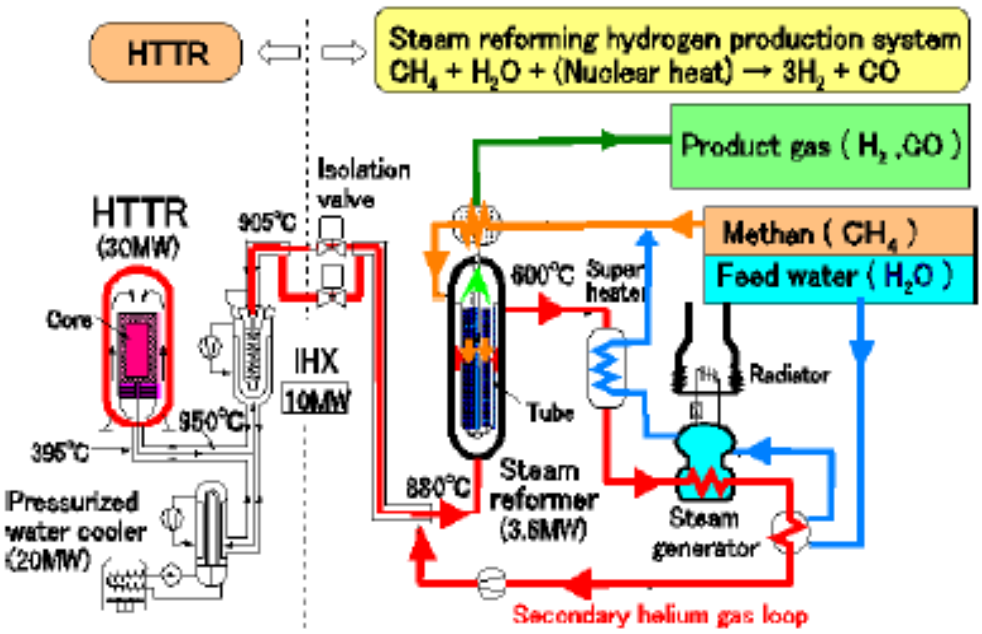


FIG. 2.26. Flow diagram of HTTR hydrogen production system.

2.4.5. HTGR development in the Republic of Korea

The Republic of Korea is much interested in future innovative nuclear energy system concepts preparing for the post-APR-1400 period. It considers within the Generation IV concept (Section 2.6.1) a nuclear hydrogen production reactor as a future innovative system. A plan to develop this GEN IV system through international collaboration has been prepared focusing on reactor system and relevant proliferation resistance fuel cycle. A nuclear hydrogen production programme for providing hydrogen economy as one of clean energy resources in the near future was started in the Republic of Korea with a reference concept of VHTR (Fig. 2.27).

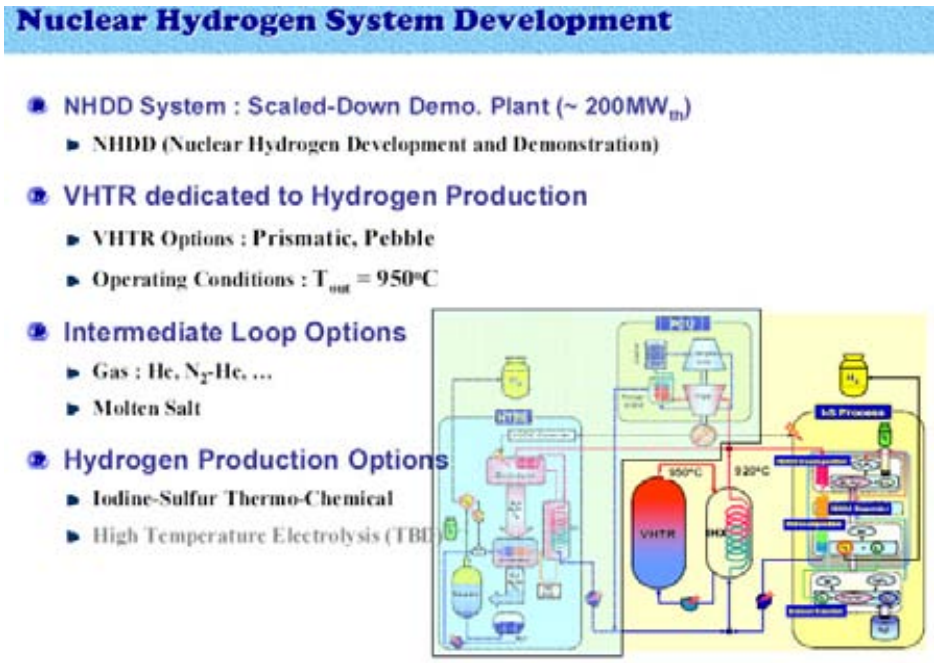


FIG. 2.27. Korean development of the VHTR concept (IAEA TWGNFCO 2006).

2.4.6. HTGR development in South Africa

A pebble bed modular reactor (PBMR) is being developed since 1993 by the consortium ESKOM in South Africa, using German expertise.

The PBMR Module power conversion is based on a closed cycle circuit, utilizing a single loop direct gas cycle system that utilizes a helium cooled and graphite-moderated nuclear core assembly as a heat source (Fig. 2.28). The coolant gas transfers heat from the core directly to the power conversion system consisting of gas turbo-machinery, a generator, gas coolers and heat exchangers. The reactor has a thermal power of 268 MW with an electrical output of 110 MW.

In 1999, a half scale model went critical at Moscow's Kurtchatov Institute demonstrating an efficiency of 45%. Control rods in the side reflector shall provide very high load flexibility.

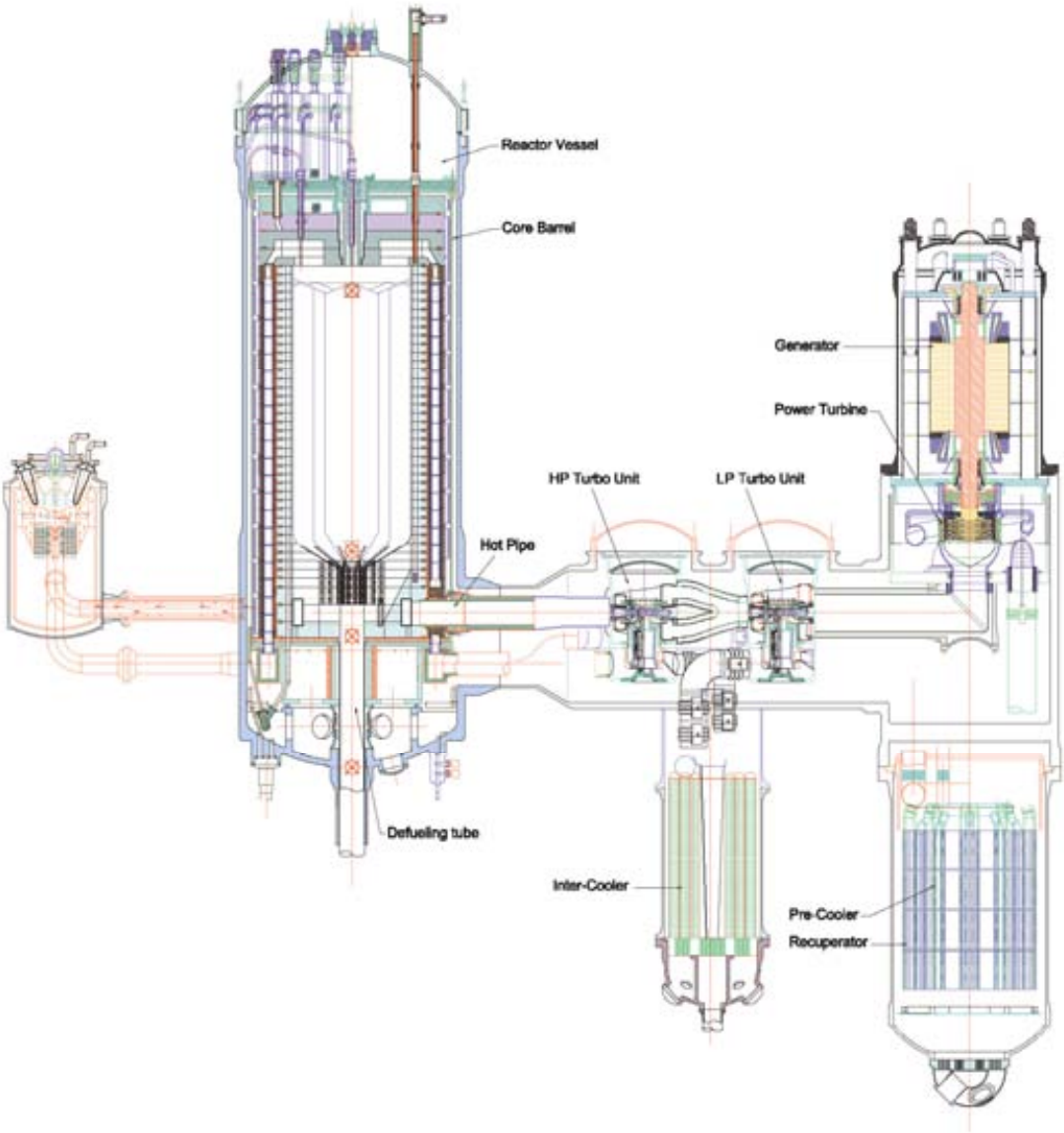


FIG. 2.28. Pebble bed modular reactor – 268 MW(th) (110 MW(e)).

The PBMR project entails the building of a demonstration reactor project at Koeberg near Cape Town and a pilot fuel plant at Pelindaba near Pretoria. The current schedule is to start construction in 2009 and for the first fuel to be loaded four years later. Construction of the

first commercial PBMR modules are planned to start three years after the first fuel has been loaded into the demonstration reactor.

Since 2004, the South African government has allocated significant funding to the project, while the Ministry of Public Enterprises stated an intent to eventually produce 4000 MW to 5000 MW of power from pebble bed reactors in South Africa. This corresponds to between 20 and 30 PBMR reactors of 165 MW each. In June 2004, the South African cabinet also approved a complimentary programme to train nuclear scientists.

The government funding enabled PBMR to secure the contracts for the development of key components such as the turbine machinery and the helium test facility. In November 2004, a contract was awarded to Mitsubishi Heavy Industries (MHI) of Japan for the Basic Design and Research and Development of the PBMR helium driven Turbo Generator System, as well as the core barrel assembly.

An important milestone was also reached on 22 November 2004, with the turning of the first sod for the construction of a helium test facility (HTF) at Pelindaba. The HTF is a high temperature, high pressure rig which will test the complete helium cycle system for the PBMR. It will also simulate the fuel-handling, reactivity control and shutdown systems.

Many countries, in particular those interested in small reactor capacities, are following this development closely.

2.4.7. HTGR development in the Russian Federation

In the 1970s–1980s the Russian Federation (OKBM) undertook substantial research on high temperature gas cooled reactors, in particular with an operating prototype reactor working at 3000 K outlet temperature.

In the 1990s, it took a lead role in the international GT-MHR (gas turbine – modular helium reactor) project based on a General Atomics (USA) design (additional partners AREVA NP (France) and Fuji (Japan)). The design was completed in 2001 (Fig. 2.29) and the prototype is to be constructed at Seversk (Tomsk, Siberian Chemical Combine) by 2010, with construction of the first 4-module power plant (4 × 285 MW(e)) by 2015. Initially it will be used to burn pure ex-weapons plutonium, and replace Pu production reactors which still supply electricity there. But in the longer term perspective HTGRs are intended to be used for burning actinides and later (via water dissociation) for hydrogen production.

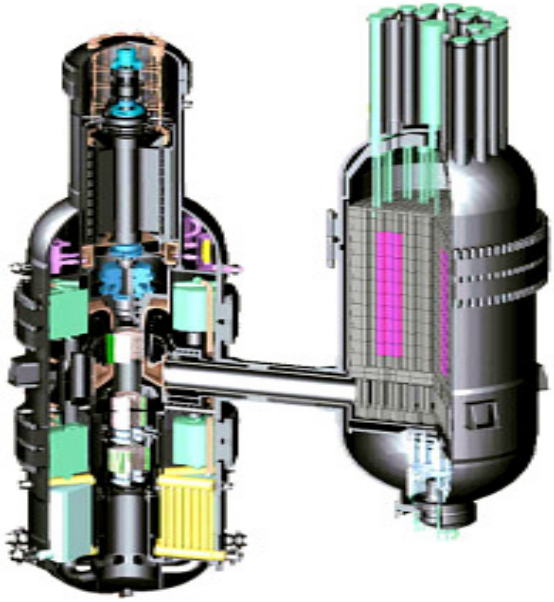


FIG. 2.29. Russian high temperature helium cooled reactor HT-MHR [27].

2.4.8. HTGR development in the USA

A gas cooled modular helium reactor with a gas turbine (GT-MHR) [28] is under development in the USA with modules of 285 MW(e) and a thermal efficiency of 48% (Fig. 2.30). As discussed above this reactor is being developed by General Atomic in partnership with the Russian Federation's ROSATOM (Ministry for Atomic Energy), and supported by AREVA NP (France) and Fuji (Japan). Its annular core consists of 102 hexagonal fuel element columns of graphite blocks with channels for the He coolant and for control rods.

GT-MHR's technical concept is based on the following advanced technologies:

- modular helium reactor, in which core destruction and melting is impossible;
- ceramic fuel design in form of small coated particles with thermo-radiation resistant coatings;
- modern technologies based on large gas turbines;
- electromagnetic bearings;
- high efficiency compact heat exchangers.

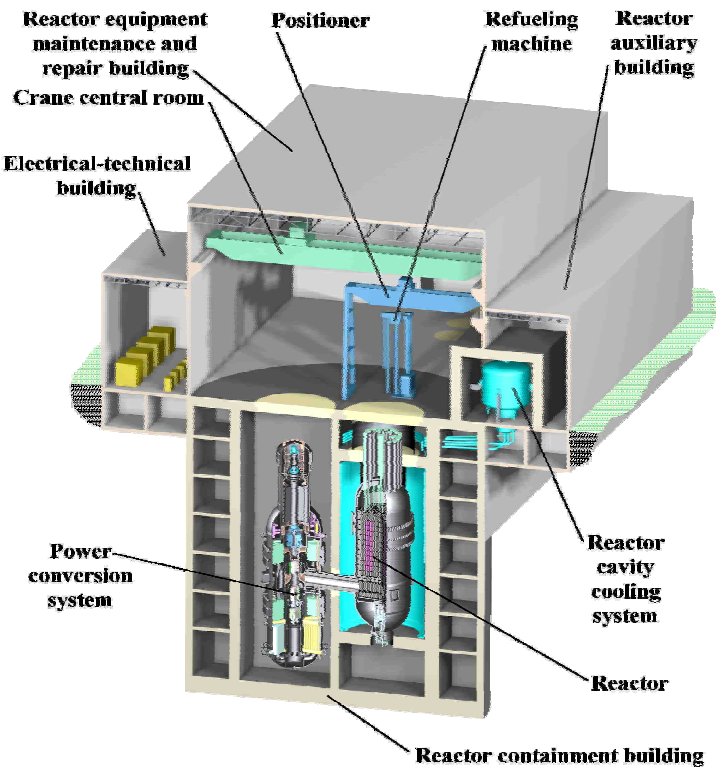


FIG. 2.30. GT-MHR module.

Regulatory review is already underway now. The Russian design is intended to be converted to US standards and to be licensed by the USNRC.

2.5. Development of fast neutron reactors (FR)

In the following section a short overview of the history of FR is given and then the situation in different countries is shortly laid out.

2.5.1. Short history of FR

The FR history is as old as that of thermal reactors. During the first 20 years of their existence, these two systems advanced side by side. The first FR was called Clementine sited at Los Alamos (USA) in 1946 with a power of 150 kW (see Table 2.11). The first nuclear reactor in the world to generate electricity was also a FR, the EBR-1 in the United States of America, in 1951. In the former Soviet Union a 100 kW FR called BR-2 started up in 1956 at the Obninsk site.

Around the sixties, four experimental fast reactors (DFR, RAPSODIE, EBR-2, and BOR-60) of about the same power went critical. The oldest and largest of these four, the DFR (72 MW(th)) in the UK, was successfully operated over 18 years; a similar operation time was achieved by the French FR RAPSODIE later on. The EBR-2 in the USA was in operation for 30 years and the BOR-60 in the Russian Federation is even still in operation now.

After DFR, which used sodium–potassium, sodium was adopted as primary coolant for FR. The first prototype fast reactor for power generation was the US Enrico Fermi reactor (EFFBR). After three years of operation, this reactor suffered a fuel melting incident and was finally shut down in 1972.

From 1972 to 1974, three prototypes of comparable size were successively brought into operation: BN-350 in the former USSR (now in Kazakhstan), Phenix in France and PFR 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. The cores of these two plants operated satisfactorily, but the plants experienced steam generator problems.

An SNR-300 prototype was built in Germany by a German-Belgian-Dutch consortium; construction of the plant and manufacturing of the MOX fuel were completed in 1985, but owing to a political impasse, the plant was never allowed to start up. The FFTF facility in the USA operated from 1985 to 1993, but it was not a power station but a large test facility[5]. The first criticality of the Japanese FR prototype Monju occurred in April 1994. Monju experienced a sodium leakage in the secondary loop in December 1995 and remains shut down till today but is scheduled to restart in 2008.

The stage of the large (pre-industrial) demonstration plants began with the startup of BN-600 (600 MW(e)) in the former USSR (Russian Federation) in 1980 (still in operation) and SUPERPHENIX (1240 MW(e)) in France in 1986 (shut down in 1996). These achievements are further discussed below in more detail, together with those in Japan and other countries. Due to technical difficulties associated with the use of sodium as a coolant and economic problems in a saturating rather than expanding nuclear energy market, BN-600 and SUPERPHENIX remained the only industrial scale fast reactors, meaning that the experience base for such reactors is much smaller than that for thermal reactors.

The nuclear breeder technology has not really become commercially up to today, at least not in an open market. On the contrary, promising advances like in France (SUPERPHENIX) were stopped in 1997 mainly due to political reasons. In the USA, this development was stopped much earlier, i.e. in 1977, by the decision of the US president Carter to stop and stay away from reprocessing irradiated fuel from commercial plants. Incidents in breeder prototypes in the United Kingdom, France and Japan, general anti-nuclear politics (especially in Germany) in combination with public opposition against nuclear reduced significantly (or even stopped) the commercial breeder programmes in these countries.

However, in the Russian Federation the fast breeder reactor (FBR) BN-600 is successfully providing electricity to the grid for 15 years since 1981 showing the best operating and production record of all Russian nuclear power plants. The BN-350 FBR operated successfully in Kazakhstan for 27 years and about half of its output was used for water desalination.

The motivation for building fast reactors has progressively changed. At the outset, the main objective for developing the FR was breeding in order to conserve uranium resources. It is easy to see the advantage of such a technology in an era of uranium shortage and price increase, as was forecast in the nineteen-seventies. 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 FRs 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 FR from the burning of plutonium to the burning (transmutation) of all transuranic actinides.

[5] In August 2006, the US DOE indicated a possible re-commissioning of FFTF as part of the GNEP programme.

TABLE 2.11. MAIN FEATURES OF FAST REACTORS IN THE WORLD

Reactor name	State	Location	First critical	Shutdown date	Thermal capacity (MW)	Electric capacity (MW)	Fuel	Primary circuit	Primary coolant	T-Primary (ºC) In/Out
Clementine	USA	Los Alamos	1946	1953	0.025		Pu metal		Mercury	140/40
EBR-1	USA	Argo (Idaho)	1951	1963	1.4	0.2	U		Na/K	
BR-2	RF	Obninsk	1956	1957	0.1		Pu metal		Mercury	70/40
BR-5	RF	Obninsk	1959	1971	5		PuO_2, UC		Sodium	450/375
DFR	UK	Dounray	1959	1977	60	15	U-Mo	Loop	Na/K	350/230
EBR-2	USA	Argo (Idaho)	1961	1991	62.5	20	U–Zr, U-Pu–Zr	Loop		482/370
EFFBR	USA	Detroit	1963	1972	200	66	U–Mo	Loop	Sodium	427/268
Rapsodie	France	Cadarache	1967	1983	20/40		MOX	Loop	Sodium	510/404
BOR-60	RF	Dimitrovgrad	1968	[b]	55	12	MOX	Loop	Sodium	550/360
SEFOR	USA	Arkansas	1969	1972	20		MOX	Loop	Sodium	430/370
BR-10	RF	Obninsk	1971	2002	8		MOX, UN	Loop		
BN-350	CIS[c]	Chevenko	1972	1999	750	150[a]	UO_2	Loop	Sodium	500/300
Phenix	France	Marcoule	1973	[b]	563	250	MOX	Pool	Sodium	552/385
PFR	UK	Dounray	1974	1994	650	270	MOX	Loop	Sodium	560/400
Joyo	Japan	Oarai	1977	[b]	50–75/100		MOX	Loop	Sodium	500/370
KNK-II	FRG	Karlsruhe	1977	1991	58	21	MOX/UO_2	Loop	Sodium	
BN-600	RF	Beloyarsk	1980	[b]	1470	600	UO_2	Pool	Sodium	550/450
FFTF	USA	Hanford	1980	1993	400		MOX	Loop	Sodium	590/370
FBTR	India	Kalpakkam	1985	[b]	40		(U, Pu)C	Loop	Sodium	518/400
Superphenix	France	Creys-Malville	1985	1997	3000	1240	MOX	Loop	Sodium	545/395
Monju	Japan	Tseruga	1994	[b]	714	280	MOX	Loop	Sodium	529/397
PFC	Italy	Brasimone	-	1990[d]	125		MOX	Loop	Sodium	525/375
SNR-300	FRG	Kalkar	-	1991[d]	770	327	MOX	Loop	Sodium	560/380
CRBR	USA	Clinch River	-	1983[d]	975	380	MOX	Loop	Sodium	
BN-800	RF	Beloyarsk	[e]		2000	800			Sodium	550/350
CEFR	China		[e]		65				Sodium	
PFBR	India		[e]		1250				Sodium	

[a] and heat for desalination [b] in operation in 2007 [c] Kazakhstan [d] abandoned [e] under construction in 2007

Table 2.12 provides a summary of experience gained with the operation of fast reactors.

TABLE 2.12. ACCUMULATED OPERATIONAL EXPERIENCE WITH FAST REACTORS

Country	Operational period
Russian Federation	110
USA	67
France	62
Japan	43
UK	38
Kazakhstan	27
India	22
Germany	14

Fast neutron reactors may be used as 'breeders', producing a surplus of ^{239}Pu from ^{238}U (breeding ration >1), or as 'burners' with just burning the produced ^{239}Pu (breeding ratio ~1). Today the main technological development goes in three directions using Na, Pb, or gas as primary coolant.

A recent IAEA report (Ref. [29]) lays out in detail information on all types of liquid metal cooled FR, such as experimental, demonstration and commercial size reactors. It includes operational parameters, physical, hydraulic and thermomechanical characteristics, technological requirements, methods and criteria to ensure safe operation; it includes design data like dimensions, materials information and main design features and performance parameters of reactor cores, components, and various systems, along with sketches and drawings.

In the following sections some illustrative examples of development activities in countries in regard to FR (liquid metal and gas cooled) are laid out.

2.5.2. FR development in China

The experimental sodium cooled fast reactor (CEFR-25) with a thermal power of 65 MW(th) and electric power of 25 MW(e) is the first step (Table 2.13) of the FR development in China (Fig. 2.31). At the end of 1993, the conceptual design was finished and is now under construction. The final safety analysis report (FSAR) is being prepared. Much progress is made on the topics required by the nuclear safety authority. The testing procedure, testing guideline and testing safety criteria for each system are under preparation. For the physics startup, the test list and test methods have been determined, and the related instruments and equipments have been ordered. Due to the delay of the delivery of the reactor main vessel, the project schedule was updated. The first criticality is postponed to the end of 2009.

Table 2.13 provides an overview of the Chinese FR programme.

TABLE 2.13. 3-STEP PROGRAMME OF FR DEVELOPMENT IN CHINA

Step	Reactor name	Power (MW(th)/MW(e))	Commissioning
1st	CEFR	65/25	2009
2nd	CPFR	1500/600	2020
Extension	CMFR	n × CPFR	On needs
3rd	CDFR	>2500/1000	2035
Extension	CCFR	>2500/1000	~2040

Legend: CEFR = China experimental fast reactor
CPFR = China prototype fast reactor
CMFR = China modular fast reactor
CDFR = China demonstration fast reactor
CCFR = China commercial fast reactor

The CEFR reactor has inherent and passive safety features, such as negative feedback of reactivity, and removal of residual heat by natural convection and natural circulation.

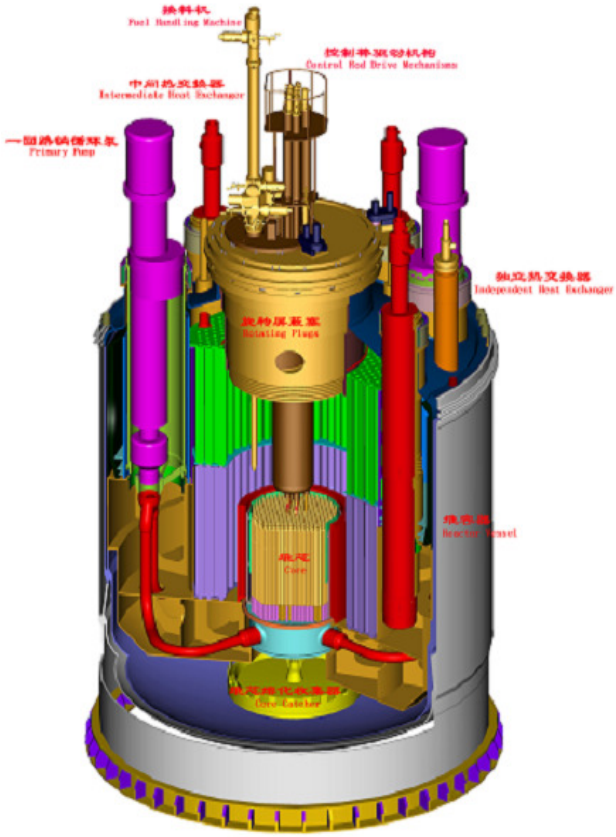


FIG. 2.31. Reactor block of CEFR.

The CPFR is proposed to the government as an important project of the Eleven's Five-Year Plan (2006–2010) with the target to complete its construction in 2020.

In order to shorten the fast reactor engineering development period from CEFR to CDFR and to decrease technical-economic risks and for their follow-up, maximum continuity will be emphasized in the main technical selections for these reactors, such as coolant, primary circuit structure, decay heat removal system, fuel handling systems, etc., as shown in Table 2.14. The demonstration fast reactor (CDFR) will be constructed around 2035.

TABLE 2.14. TECHNICAL CONTINUITY OF CHINESE FBR DEVELOPMENT

	CEFR	CPFR	CDFR
Power, MW(e)	25	600	1000～1500
Coolant	Na	Na	Na
Type	Pool	Pool	Pool
Fuel	UO_2 MOX	MOX Metal	MOX Metal
Cladding	Cr–Ni	Cr–Ni, ODS	Cr–Ni, ODS
Core Outlet Temp, °C	530	500～550	500
Linear Power, W/cm	430	450～480	450

TABLE 2.14. TECHNICAL CONTINUITY OF CHINESE FBR DEVELOPMENT (cont.)

	CEFR	CPFR	CDFR
Burnup, MW·d/kg	60～100	100～120	120～150
Fuel Handling	DRPs SMHM	DRPs SMHM	DRPs SMHM
Spent Fuel Storage	IVPS WPSS	IVPS WPSS	IVPS WPSS
Safety	ASDS PDHRS	ASDS + PSDS PDHRS	ASDS + PSDS PDHRS

DRPs: Double rotating plugs
ASDS: Active shutdown system
SMHM: Straight moving handling machine
PSDS: Passive shutdown system
IVPS: In-vessel preliminary storage
PDHRS: Passive decay heat removal system
WPSS: Water pool secondary storage

2.5.3. FR development in France

In France, the targeted objectives for future systems as well as the choice of key technologies necessary in achieving them have become the object of very active international cooperation, particularly within the framework of the Generation IV International Forum (GIF, to be discussed later in Section 2.6.1).

The French interest into the nuclear future [24] goes into 4 directions (Fig. 2.32).

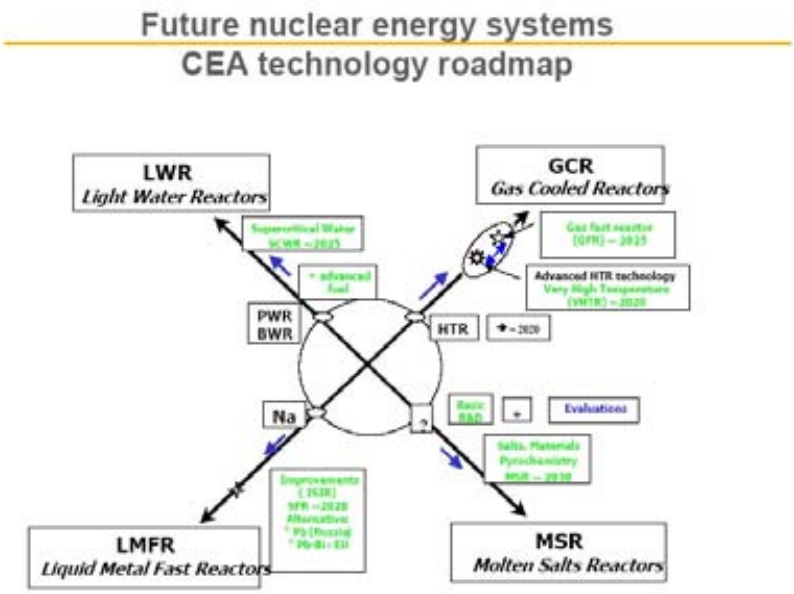


FIG. 2.32. Future French nuclear energy systems.

Gas cooled fast reactor

Guidelines have been drafted to implement an R&D programme to establish the viability of a gas cooled fast reactor (GFR) and to complete a conceptual design. The guidelines include consideration of an experimental GFR of limited power (the Experimental Technology Demonstration Reactor, ETDR) to help taking the decisions of building a prototype GFR system to be placed into operation by 2035 (Fig. 2.33).

The ETDR will be the first GFR ever built. It will be a low power experimental reactor (20 to 50 MW(th)) to demonstrate the viability of the specific technologies of GFR (e.g. fuel, safety systems) and also to bring elements of demonstration for the whole gas cooled reactors technological pathway. Considering a start of operation in 2015, it will be an essential step in the decisions to be taken by 2019 for launching a prototype GFR (Figs 2.34 and 2.35) system to be placed into operation by 2035. As such, it intervenes in particular in the fuel development plan in between the irradiation phase of materials and fuel samples in MTRs and available fast reactors prototypes and the full scale demonstration phase in the prototype GFR of which the ETDR has to be considered as a precursor.

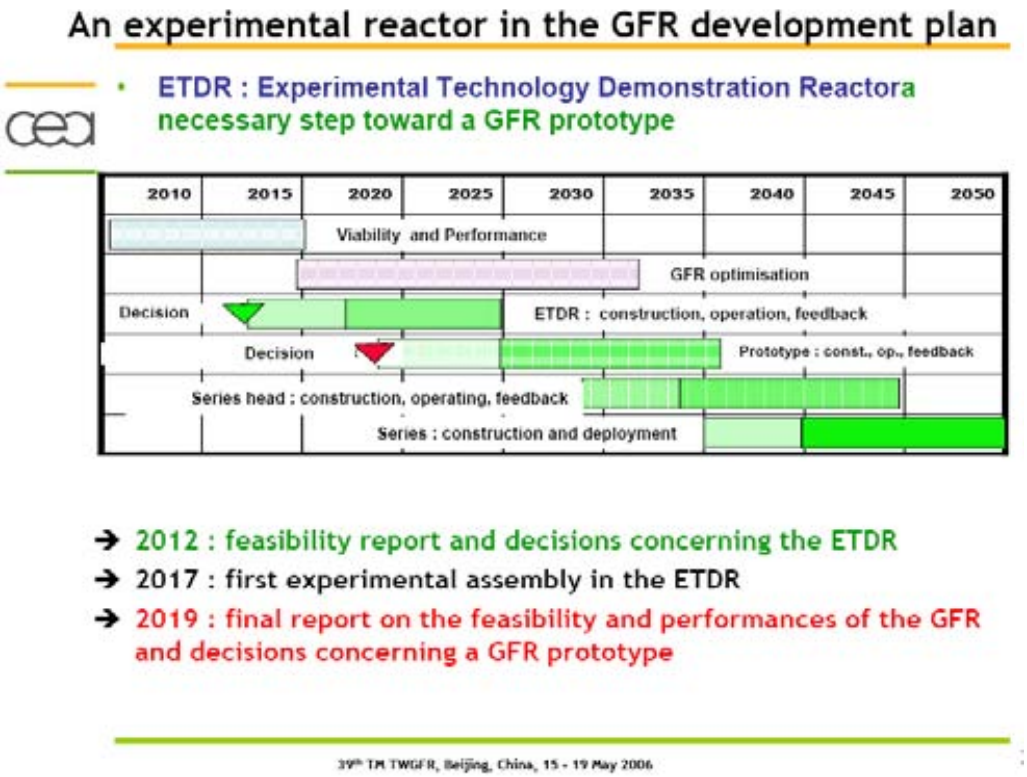


FIG. 2.33. The French development programme for GFR.

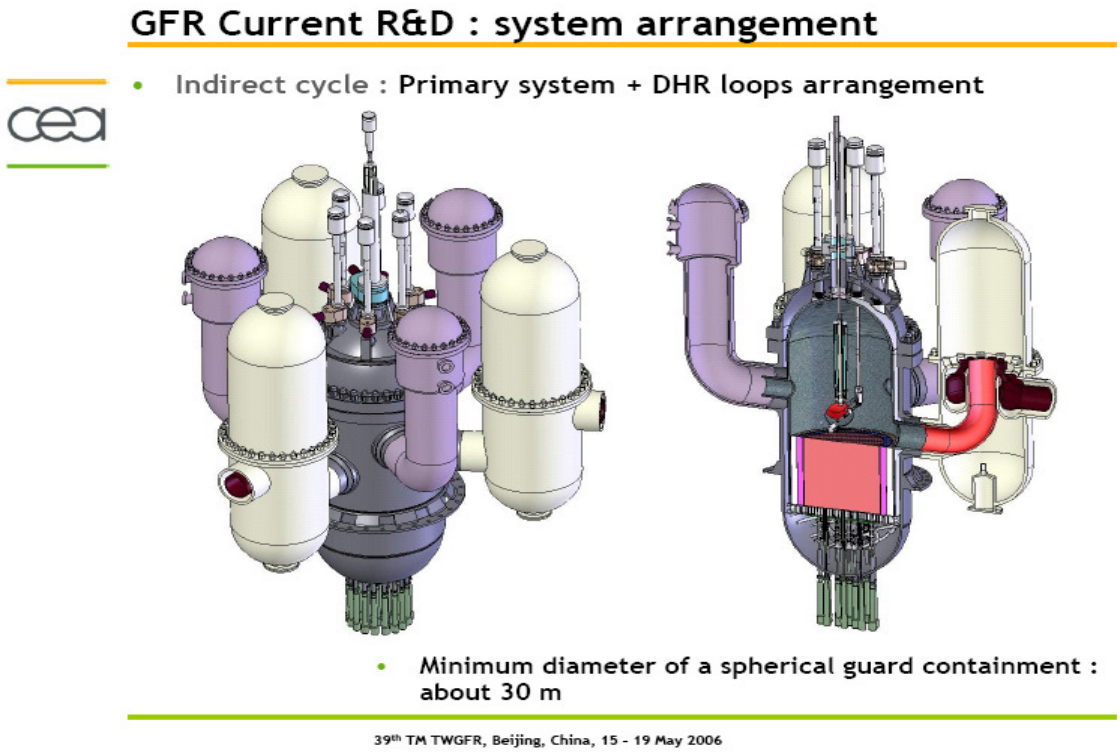


FIG. 2.34. French gas cooled fast reactor design.

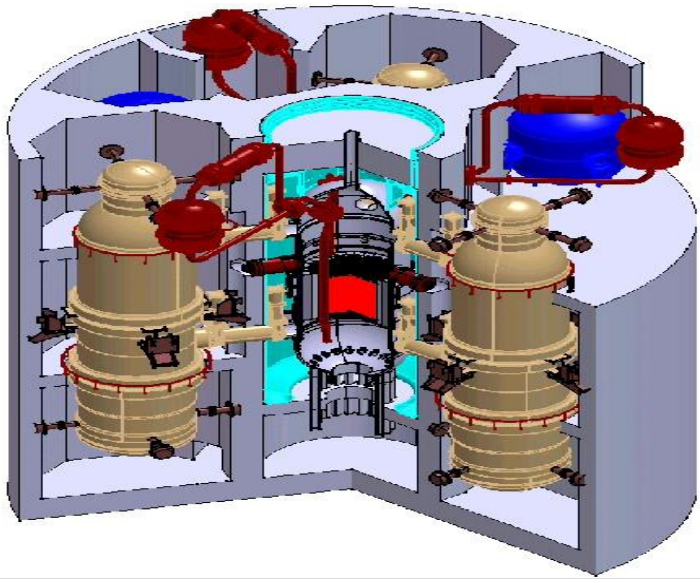


FIG. 2.35. Containment of the French gas cooled fast reactor (GFR).

Common technologies for materials and components can be developed in line with the VHTR (considered in GIF) dedicated to hydrogen production (Section 2.6.1). But anyway there is a need for a large R&D programme enabling both a fast spectrum and high temperature.

The main characteristics of the GFR are an attractive power density (~100 MW/m^3), self-generating (i.e. with a high conversion ratio) cores with a fast neutron spectrum, robust refractory fuel, a high operating temperature, direct energy conversion with a gas turbine, full actinide recycling possibly associated with an integrated on-site fuel reprocessing facility, high level of fission products confinement, and increased resistance to severe accidents. Additionally, the coolant (gas) does not undergo a phase change (as e.g. in water cooled reactors), it is chemically inert and optical transparent.

Sodium cooled fast reactor (SFR)

In France, there is one sodium cooled FR in operation, called PHENIX, mainly for experimental purposes. PHENIX went critical in 1973 and reached full power in 1974, after a rapid construction period (1968–1973) without difficulties.

PHENIX is a pool-type reactor; the primary circuit is enclosed in the main vessel reactor (11.8 m diameter), filled with 800 t of sodium, containing three primary mechanical pumps and six intermediate heat exchangers (IHX) connected to three independent secondary loops. Each secondary loop contains a steam generator (SG) consisting of three stages: evaporator, super-heater and re-heater, twelve modules each. Between 1974 and 1990, the plant was used for irradiation experiments and electricity production (250 MW(e) electrical output), with a very satisfactory operational record for a prototype (51 cycles, 61% average load rate and more than 20 billion kW·h produced).

During these 16 years of operation, mainly two periods of unsettled operation (with shutdown or at 2/3 of nominal power) occurred in order to repair and modify IHX (1976–78) and SG (1982–83). In addition some very limited sodium fires occurred due to small sodium leaks (few kg) in the secondary circuit. Safety analyses of these incidents showed that they could be seen as a possible common mode failure of the secondary circuits. For future reactors, an effective separation of the circuits has been asked for as a safety criterion. In 1989 and 1990, four emergency shutdowns were activated by negative reactivity signals; analysis of the origin and research into the potential consequences required a long development. A complete review of the different scenarios of reactivity change was set and renewed for SUPERPHENIX (similar scenarios could not jeopardize the SUPERPHENIX safety).

The development of sodium cooled fast reactors in France is shown in Fig. 2.36.

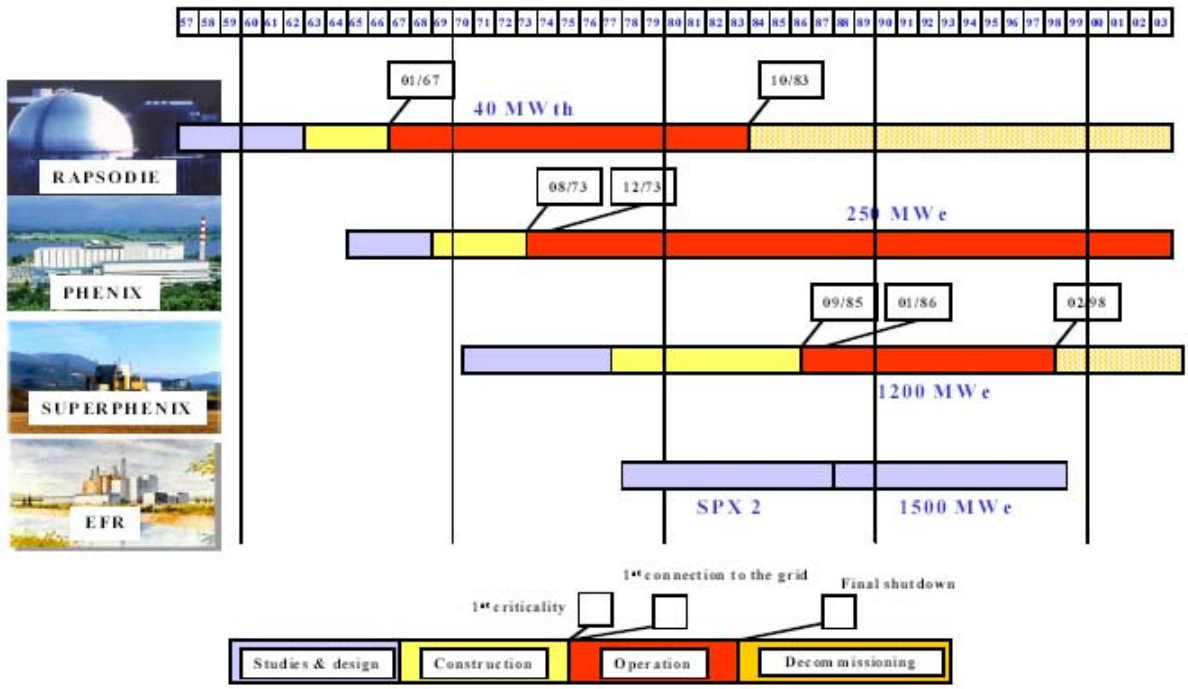


FIG. 2.36. French development of Na cooled FR.

In the mid-1990s, the role of the PHENIX reactor changed: it was to be used as an irradiation tool acting as a support to the R&D transmutation programme of the CEA within the framework of the 1991 French law concerning long lived radioactive waste management. As

the initial lifetime of the reactor was 20 years, the reactor should have been shut down in 1994, but this new objective required an extension of the planned reactor lifetime. Therefore, a large renovation programme was defined and performed till February 2003. Resuming power happened in June 2003: six operating cycles are foreseen, covering about five years and a half, to carry out the irradiation experimental programme on minor actinides and long lived fission products transmutation and on future reactor options.

From the beginning, PHENIX appeared as a reactor quite easy to control and operate. Similar to RAPSODIE, it was a very efficient irradiation facility thanks in particular to an adjacent hot cell allowing intermediate examination of pins and return to the reactor. Records of burnup on oxide and dose on materials (155 dpa) were achieved and contributed to the optimisation of fast reactor fuels.

But also in the fields of physics and sodium technology, the contribution of PHENIX is considerable: as an example, it was the first reactor to confirm the breeding capacity of a fast reactor, by recycling Pu from its own previous core elements; it demonstrated also, at several occasions, the possibility to remove big components (e.g. intermediate heat exchangers or primary pumps), to clean and repair them, then to reintroduce them in reactor for further operation.

The sodium cooled fast reactor (SFR) is one of the systems selected by the roadmap of the Generation IV International Forum (Section 2.6.1), and the CEA is engaged in a substantial effort in the viability phase. The considerable amount of experience accumulated in France over the past 40 years of R&D and feedback experiments will be used but some improvements are needed on the SFR to meet the GEN IV goals. A set of four projects is under preparation to meet the corresponding objectives:

- design and safety;
- advanced fuels and materials;
- component design and BOP;
- fuel cycles (front end and back end).

The CEA will participate in the four projects, providing both significant expertise and original work.

2.5.4. FR development in India

India's roadmap for development of their nuclear energy system was discussed shortly in Section 2.3.5.

In the first step of its liquid metal fast breeder reactor (LMFBR) programme, India could leap frog and introduce an advanced mixed uranium plutonium mono-carbide driver fuel, of hitherto untried plutonium rich composition in the FBTR at Indira Gandhi Centre for Atomic Research (IGCAR).

FBTR is a 40 MW(th)/13.2 MW(e), mixed-carbide fuelled, sodium cooled, loop type reactor with two primary and two secondary sodium loops. Each secondary loop has two once-through serpentine type steam generators (SG). A turbogenerator (TG) and a 100% steam dump condenser (DC), to facilitate reactor operation without TG, are also provided. The first criticality of the reactor was achieved in October 1985 with a small core of 22 fuel subassemblies (SA) of MK-I composition (70%PuC + 30%UC), with a design power of 10.6 MW(th) and peak linear heat rating (LHR) of 250 W/cm. The TG was synchronized to the grid in July 1997 for the first time.

The core has since been progressively enlarged by adding SA at the peripheral locations. The peak LHR of the MK-I had been raised to 400 W/cm, and the reactor has been operated up to a maximum of 17.4 MW(th). Several irradiation campaigns have been completed. In 2004, the core had 39 fuel subassemblies; the reactor had logged 1374 d of cumulative operation of which 704 d has been at high power with SG in service; the total thermal energy generated is 208 GWh; TG had operated for a cumulative of 4173 hrs, generating 3.948 million units of electricity; and the carbide driver fuel (70% PuC + 30% UC) in the first ring has accumulated a burnup of 123.43 GW·d/t without clad failure. A few irradiated low burnup (25000 MW·d/t) MC fuel pins from FBTR have been successfully reprocessed.

On 18 July 2006, the fast breeder test reactor (FBTR) completed 20 years of its successful operation at Kalpakkam. The 13th irradiation campaign with 43 fuel subassemblies was completed. During this campaign, the reactor was operated at high power (16 MW(th)) and logged a continuous run of 37 days. Four Mark-I subassemblies loaded also accumulated 155 GW·d/t burnup. The eddy current flow meter was installed and sodium flow through subassemblies was measured as a prerequisite for getting safety clearance for the next irradiation campaign. SARCOP clearance was obtained for the 14th irradiation campaign. Documents, including the safety report for the Hybrid Core, were submitted for periodic safety review by AERB.

Based on the experience gained with this reactor and with cooperation of research centres and industry, IGCAR has completed detailed design and technology development of the 500 MW(e) prototype fast breeder reactor (PFBR 500) with mixed uranium plutonium oxide as reference fuel (Fig. 2.37). Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI), a government company was incorporated on 22 October 2003 at Chennai, Tamilnadu, for implementing the India's first commercial fast breeder reactor project.

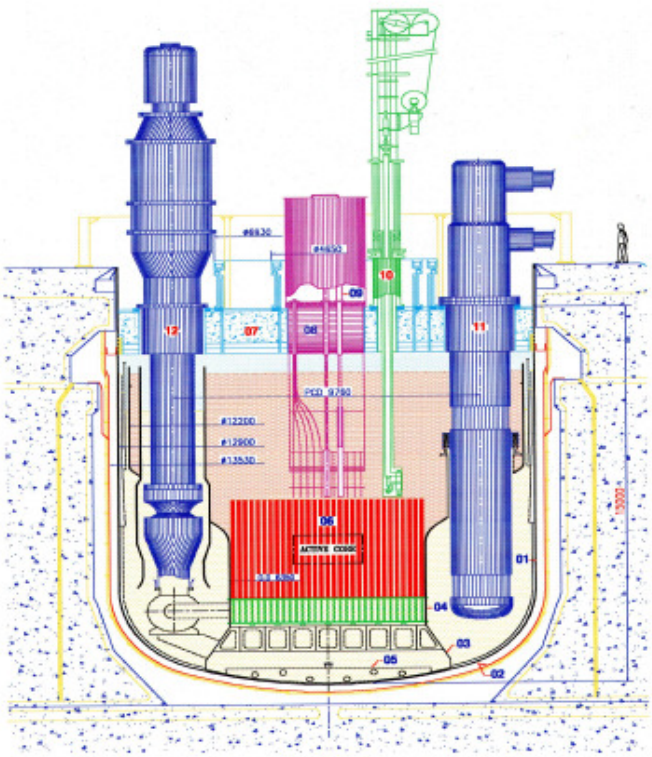


FIG. 2.37. Sketch of the 500 MW(e) PFBR.

In 2006–2007 the civil construction of Nuclear Island buildings for PFBR made progress. The construction of the outer peripheral wall around nuclear island connected buildings (NICB) raft, which acts as a retaining wall with counter forts for retaining the earth, was completed. The construction of super structures on the NICB raft commenced and progressed at various elevations. The construction of electrical building 1&2 and control building was completed up to operating floor level. Construction of the service building was completed up to roof level.

Important works planned include the PFBR shielding experiment, testing of a transfer arm in air, boron enrichment, post-irradiation examination of FBTR fuel subassembly after a burnup of 100 GW·d/t, structural integrity testing and reprocessing of FBTR carbide fuel.

A concept of a liquid metal cooled reactor (although not a FR) is being developed in India called compact high temperature reactor (Fig. 2.38). It has the following features:

- passive core heat removal by natural circulation of liquid heavy metal coolant;
- passive power regulation and shut down mechanism;
- passive rejection of entire heat to the environment under accident condition.

The background for this system is the use of Th-fuel.

This reactor system roadmap can only be fully understood on the background of the fuel cycle concept in India, which was discussed shortly in Section 2.3.5.

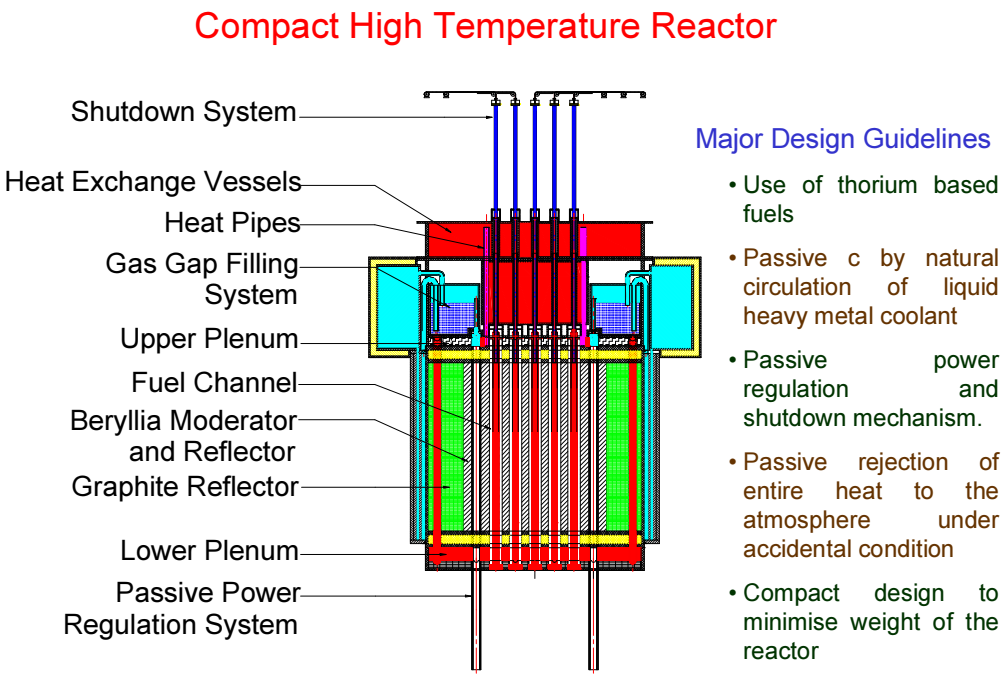


FIG. 2.38. Schematic of compact high temperature reactor (India).

2.5.5. FR development in Japan

In Japan many studies are being performed [30] in many different potential technological directions. However, the focus of efforts appears still to be the fast reactor systems (Refs [31, 32]) considering all today known coolant technologies (Na, Pb + Bi, He and H_2O), as well as all kinds of today known U and Pu fuel materials. There are large, medium and small sizes of reactors investigated.

Sodium cooled fast reactor

The history of the development of sodium cooled FR in Japan is shown in Fig. 2.39.

The experimental fast reactor JOYO has been supporting the development of sodium cooled fast reactors by providing valuable irradiation testing of advanced fuels and materials and improvements in fast reactor safety and operation since criticality was first achieved in 1977 as the MK-I breeder core. The first major upgrade of JOYO to the 100 MW(th) MK-II irradiation-test-bed was successfully operated from 1982 to 2000.

Work began in 2000 on the 140 MW(th) MK-III programme, which was the second major upgrade to enhance the irradiation capability of JOYO. For the MK-III programme, the entire core and major heat transport components were redesigned and replaced to provide increased neutron flux and irradiation testing capability.

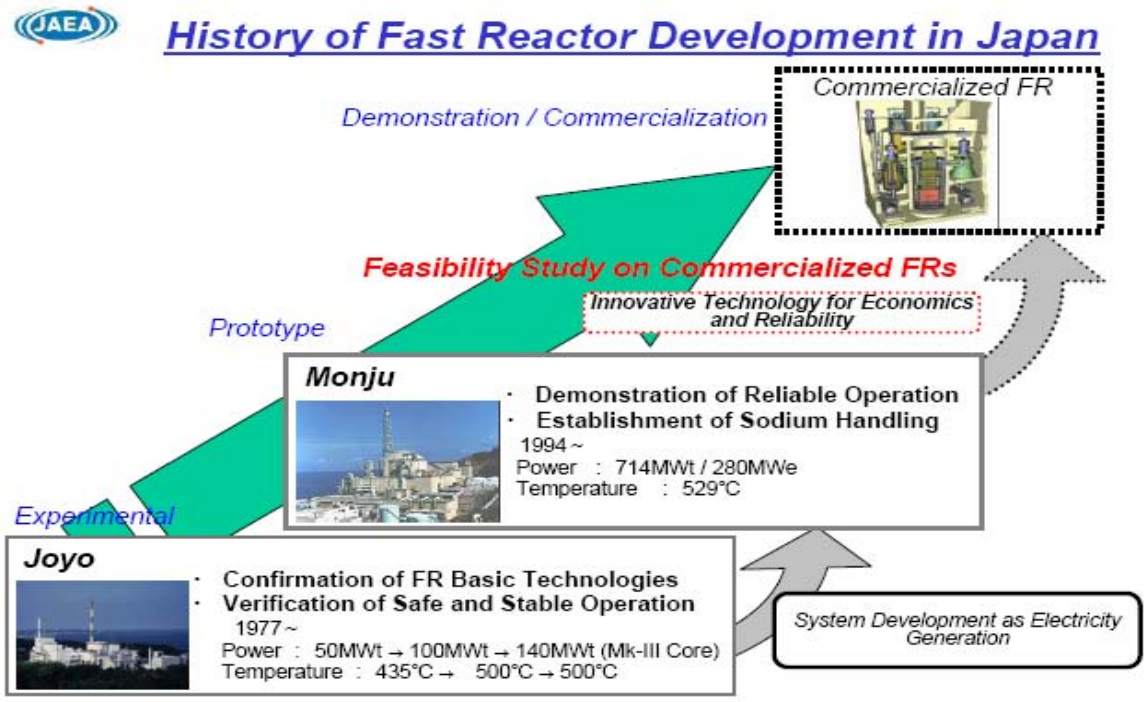


FIG. 2.39. History of sodium cooled FR development in Japan.

Functional and performance testing verified the design parameters of the JOYO MK-III core. The pre-service inspection by MEXT was completed on 27 November 2003. A utilization plan for future fuels and materials development and safety testing in the JOYO MK-III core has been developed.

The prototype FBR MONJU (280 MW(e)) reached first criticality in 1994 and started generation of electricity in 1995. Since that time it is under pre-operation test conditions to master the power plant technologies, improve performance and establish economic efficiency. However, the pre-operational test of the plant was abruptly interrupted by a sodium leak accident in the secondary heat transport system in December 1995 during a 40% power operation test. After carrying out the root cause investigation and the comprehensive safety review for two years and the necessary licensing procedure, the permit for plant modification (countermeasures against potential sodium leaks, etc.) was issued in December 2002 by the Ministry of Economy, Trade and Industry (METI).

JAEA has started preparatory work for modification, after given prior approval by the local governor of Fukui in February 2005, and the main modification work is in progress since September 2005, and has achieved 63% completion by the end of April 2006. After

modification work, the functional tests will be carried out for the modified systems and for the system as a whole, especially considering the long period of plant shutdown, and the required refuelling. Criticality is expected to be achieved early in 2008.

In January 2003, the Kanazawa Branch of Nagoya High Court approved the local resident's appeal which requires the nullification of government's licence on MONJU installation. The lawsuit was then brought up to the Supreme Court, which finally accepted the government's appeal on 30 May 2005. This means the lawsuit against MONJU fought for more than 20 years is now over and there is no legal obstacle for MONJU operation any longer.

Currently the R&D operation plan for MONJU after restart is planned in the two stages. In the first stage of MONJU operation, it is expected to demonstrate its reliability as a FR power plant and establish sodium handling technologies. The operational data from MONJU will be used to assess and validate the design of the core, plant systems and components.

In the second stage, MONJU should be used to verify the elemental technologies towards commercialized FR cycle systems, including the demonstration of burning minor actinides in FRs to reduce the environmental burden caused by high level radioactive wastes. At the same time, MONJU can be used as a valuable tool for the international cooperation. The long term R&D programme includes demonstration tests of a long-life and high burnup advanced fuel, MA bearing fuel, etc.

JNC has performed a feasibility study [33] on commercializing a FR cycle system with participation of all parties concerned in Japan, i.e. the electric utilities, Japan Atomic Energy Agency (JAEA), Central Research Institute of Electric Power Industry (CRIEPI), and Japanese manufacturers since July 1999. The feasibility study had the aim to define the most promising FR system technologies (Fig. 2.40) available for commercialization by around 2015. Such a FR system could supply sustainable energy both to allow long term use of nuclear power by recycling TRU fuel and to have a potential to be harmonized with the environment by burning miner actinides (MA) and transmuting long-life fission products (LLFP).

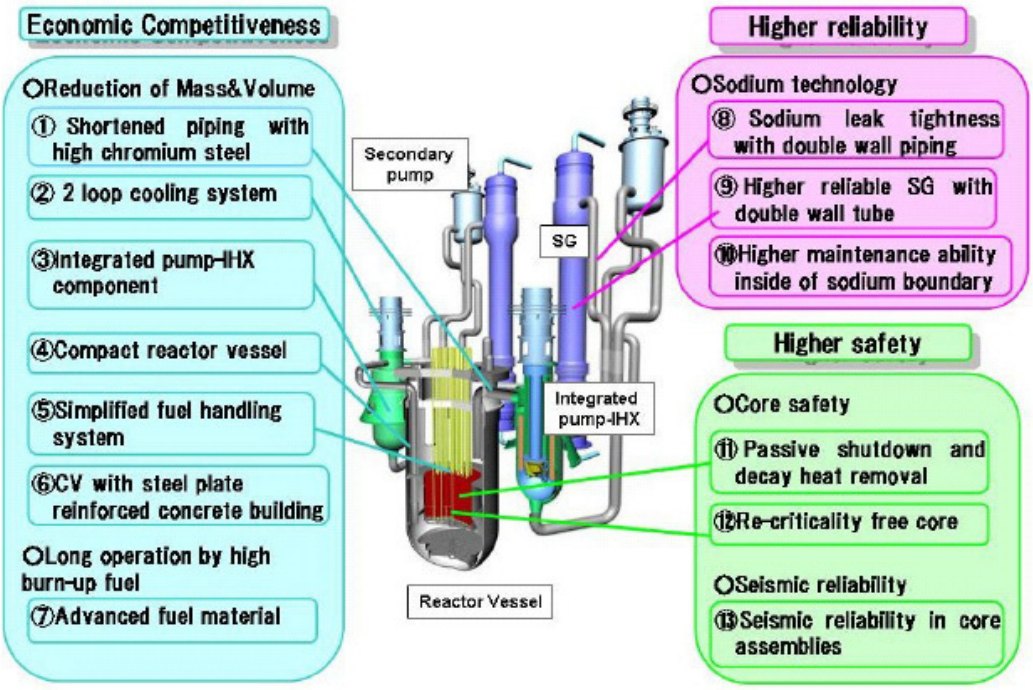


FIG. 2.40. Design improvements of the Japanese sodium cooled fast reactor (JSFR).

An advanced fuel cycle system selected in the feasibility study will be demonstrated in engineering scale R&D facilities, such as the recycle equipment test facility (RETF) in Tokai Works by using spent fuel from JOYO and MONJU in JNC. These sequential approaches will address the optimization of the FR fuel cycle system by the year 2015 (Fig. 2.41).

As for the feasibility study on commercialized fast reactor cycle systems, a three-year period from JFY 2001 to 2003 was the initial term of phase II. During this term, research activities were focused on the design of the candidate concepts and on the fundamental tests of key technologies.

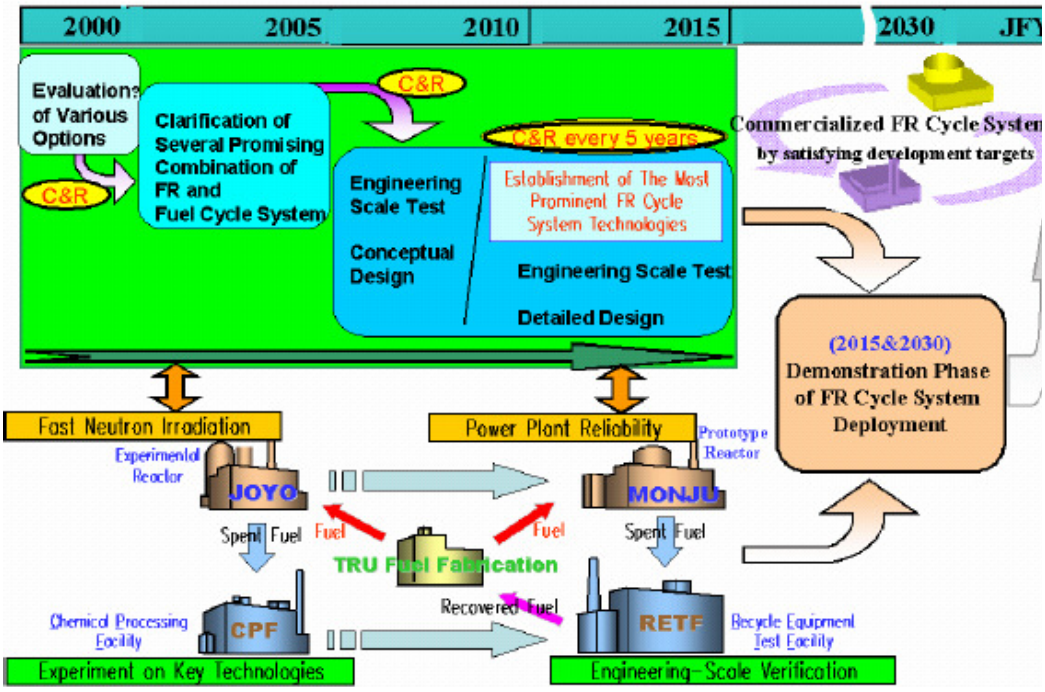


FIG. 2.41. Development steps of the NFC for FR in Japan.

The interim evaluation has been completed for the combinations of several of these concepts, and the feasible candidate concepts are clarified. Also crucial R&D items have been identified and summarized as a roadmap of the key technology development for each candidate (Table 2.15).

TABLE 2.15. EVALUATION OF ACHIEVEMENT TO FR DESIGN REQUIREMENTS

Items		Targets	Sodium-cooled FR (750, 1500MWe) MOX	Sodium-cooled FR (750, 1500MWe) Metal	LBE-cooled FR (770MWe) MN Fuel	Helium-cooled FR (1124MWe) MN Fuel	Water-cooled Reactor (1356MWe) MOX Fuel
Burn-up	averaged over core	>150GWd/t	150GWd/t	150GWd/t	150GWd/t	120 GWd/t	90 GWd/t
	averaged over core + blanket	>60GWd/t	60(a)-110(b) GWd/t	60(a)-145(b) GWd/t	105(a)-140(b) GWd/t	60(a)-80(b) GWd/t	45 GWd/t
Breeding ratio		about 1.0~1.2	1.04(b) - 1.16(a)	1.03(b)- 1.17(a)	1.04(b) - 1.15(a)	1.04(b) - 1.17(a)	1.03*
Compound System Doubling Time		-	50 years	40 years	50years	70 years	>200 years
Pu Fissile Inventory		-	4.5(a) - 6(b) t/GWe	4(a) - 7(b) t/GWe	6 t/GWe	10 t/GWe	≅11 t/GWe
Operation cycle length		>18 months	18(a) - 26(b) months	20(a) - 28(b) months	18 months	20 months	18 months
Primary coolant temperature		-	550 / 395°C	550 / 395°C	445 / 285°C	850 / 460°C	287 / 285°C
Thermal efficiency		-	42%	42%	38%	47%	35%
Construction cost		<100 points	91(L) - 94(M) points		108 points	110 points	≅100 points

* Total Surviving Ratio of Pu Fissile

(a) Core of high breeding ratio, (b) Core of low fuel cycle cost

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As the base load power source, a large scale FR (1500 MW(e)) and a medium scale modular FR (750 MW(e)) are under investigation (Fig. 2.42). The former is pursued to reach the lower construction cost by using the economy of scale merit, and the latter is pursued to reduce the cost by modular construction.

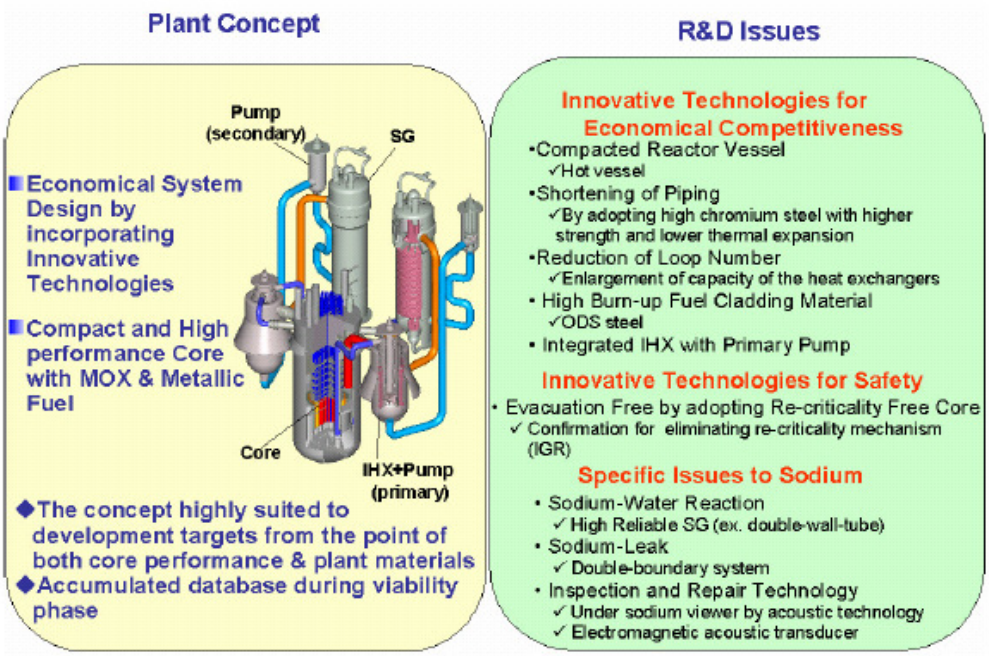


FIG. 2.42. Current status of Japanese design study on sodium cooled FR system.

The feasibility study on commercialized fast reactor cycle systems was finalized in 2006 with the selection of the sodium cooled fast reactor with oxide fuel, an advanced aqueous reprocessing and the simplified palletizing fuel fabrication process. As a follow up project the fast reactor cycle technology development project (FaCT, Ref. [34]) was started.

SMR

In addition to the concepts of large size FR also modular small and medium reactors (SMR) with sodium cooling and a fast neutron spectrum are studied. An example is the reactor MDP with 325 MW(e) [11], and two other ones called RAPID (1 MW(e)) and 4S (10 or 50 MW(e)) that do not require on-site refuelling [12].

Lead–bismuth eutectic (LBE) cooled fast reactor

A comparative design study is performed on a medium scaled modular FR (750 MW(e)) with nitride fuel. In this concept, several advanced technologies using the characteristic of the LBE have been adopted to achieve the development targets (Fig. 2.43).

In the design of a LBE cooled FR, the fuel pin cladding temperature and flow rate at fuel bundle are limited to 650°C and 2 m/s respectively to control LBE corrosion. Therefore, an LBE cooled FR could not achieve the high coolant temperatures as expected from LBE boiling point (1670°C) and nitride fuel melting point (2750°C). But reactor core performance still would have a potential to attain the development targets on condition that the safety objectives are obtained.

By applying nitride fuel, the lead–bismuth cooled reactor has the potential to achieve core performance equivalent to the sodium cooled reactor and meet all the design requirements. Concerning technical feasibility, essential issues include the corrosion of steel such as the fuel cladding in addition to the development of the nitride fuel. Accordingly, fundamental R&D is

needed to develop corrosion prevention technology and corrosion resistant material, which will determine the conceptual applicability.

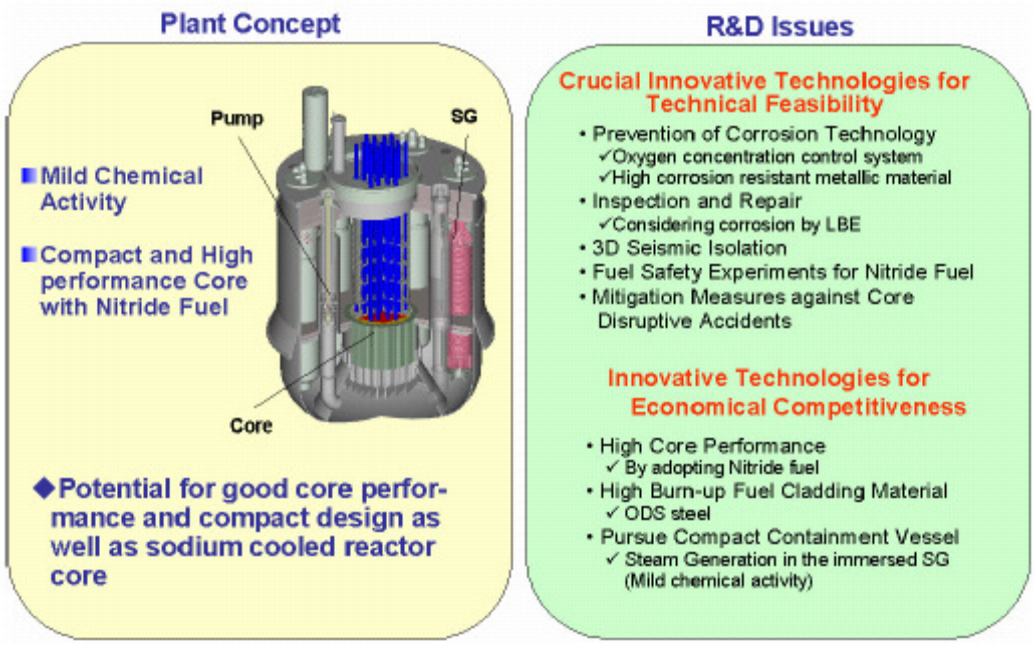


FIG. 2.43. Design study on Japanese lead–bismuth cooled FR system.

Although the lead–bismuth cooled reactor was also selected as one of the candidate reactor types at the GIF project, originally no country has taken leadership in its development thus far, but it might be possible that the new member (as of 2006) of GIF, the Russian Federation, might take the lead in this area.

Helium cooled coated-particle-type fuel fast reactor

A helium gas cooled FR (1124 MW(e)) has been examined with high thermal efficiency by achieving high core outlet temperatures (up to 850°C) and introduction of a compact gas turbine system and coated particle fuel concept. Reactor core performance in accordance with design requirements would be attained by introducing nitride coated particle fuel for both high and low breeding ratios with high burnup of TRU fuels. However, it was found that the core performance is lower than a sodium or LBE cooled reactor. Nevertheless the high temperature gas cooled FR has the attractive feature as a high temperature heat source, which can be used for multipurposes such as hydrogen generation (Fig. 2.44). It can be defined as a promising concept, if the related fuel materials and subassemblies can be developed (Section 4.6.5).

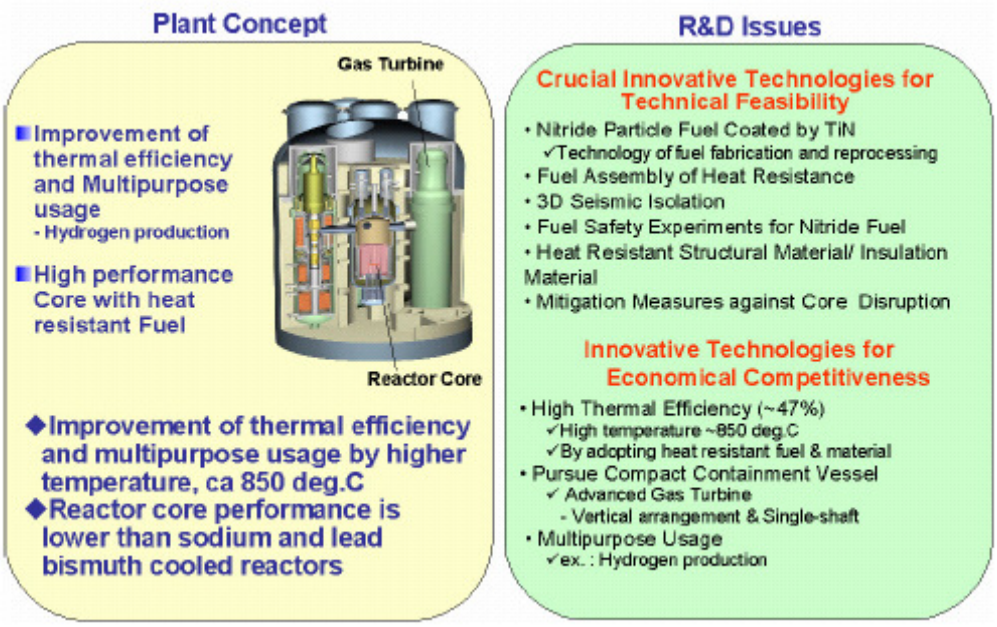


FIG. 2.44. Status of Japanese design study on helium gas cooled FR system.

Water cooled fast reactor

Boiling water type and pressurized water type FR have been examined by JAEA (Fig. 2.45), and a supercritical water type (SCFBR) has also been studied by Tokyo University. Furthermore, the issues with core disruptive accidents of SCFBR have been examined in collaboration with FZK/Germany. Hereafter, BWR type reduced moderation water reactor concept is described because the design work of BWR type has made more progress comparing to other types at this stage.

The water cooled FR concept is expected to use LWR operational experiences especially in service inspection and repair (ISI&R) aspects, and thus the plant reliability should be higher than other FR concepts.

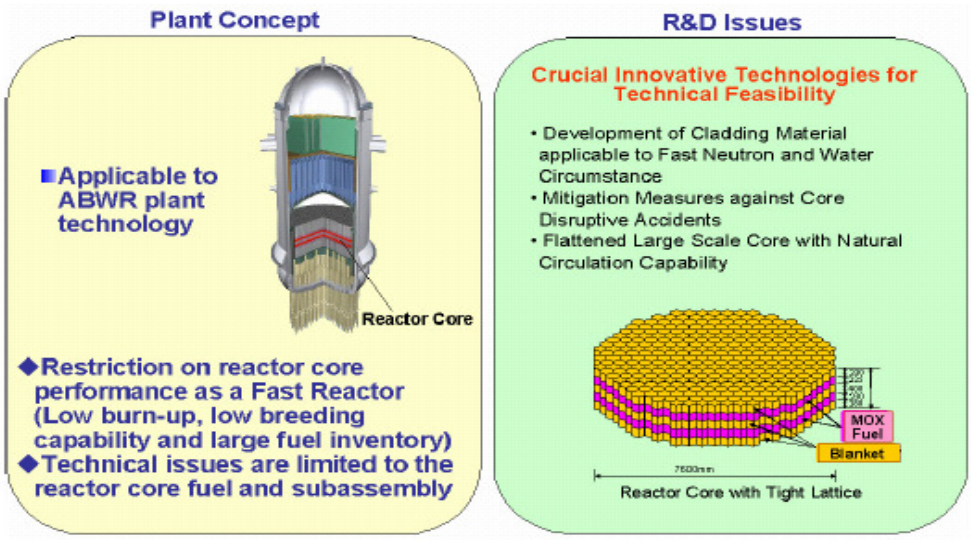


FIG. 2.45. Design study on Japanese water cooled FR system (R&D of JAEA).

On the other hand, the specific FR issue, also common to the water cooled concept, is the FR core performance. Namely the breeding ratio and the burnup is limited even when highly

decontaminated U/Pu fuel is used, and also the fuel inventory tends to be larger, which would increase the fuel cycle cost. When low decontaminated TRU fuel is used, the core performance would become worse due to the fact that the void reactivity has to be negative in the core and the fuel enrichment is required to be under an allowable level from viewpoints of fuel fabrication and reprocessing. In addition, the water cooled reactor has lower performance in accepting and burning minor actinides (MAs) that are recovered by the reprocessing of spent LWR fuel, compared with other reactor concepts. It has the potential of meeting the other design requirements such as safety, economy, and nuclear proliferation resistance.

In order to proof its technical feasibility, the water cooled reactor concept has difficulties, which are, however, limited to fuel-related issues. It is necessary to develop cladding material and to discuss countermeasures for the mitigation of the consequences of core damage. In addition, since a boiling water reactor (BWR)-type FR, which was discussed in this study, was not selected as a candidate reactor type at the GIF project, international cooperation is limited to basic research topics at this time.

2.5.6. FR development in the Republic of Korea

Future nuclear power plants should meet several challenges: effective resource utilization, minimized waste and reduced environmental impact, economic competitiveness, enhancement of safety and reliability, proliferation resistance and physical protection.

In order to meet these challenges the Ministry of Science and Technology (MOST) established the Comprehensive Nuclear Energy Promotion Plan (CNEPP) of 2001 which sets the basic framework of nuclear R&D. CNEPP proposed developing liquid metal reactor technologies for the efficient utilization of uranium resources with an emphasis on basic key technologies; and CNEPP also suggested participating in international collaborations including the GEN IV programme. Furthermore, as a result of Nuclear Technology Roadmap activities in the Republic of Korea, the sodium cooled fast reactor was chosen to be one of the two future types which should be deployable by 2030.

The fast reactor technology development project is funded by the MOST under the National Medium and Long term Nuclear R&D Programme. The objective of the project is to develop basic key technologies for the liquid metal reactors which can meet the goals of sustainability, safety and economic competitiveness. The work scope of the project includes reactor design studies, development of computational tools, and development of sodium technologies.

Sodium cooled fast reactors

The sodium cooled fast reactor technology development project has been carried out as shown in Fig. 2.46. During Phases 1&2, basic technologies and the conceptual design of KALIMER-150 with 150 MW(e) capacity has been developed.

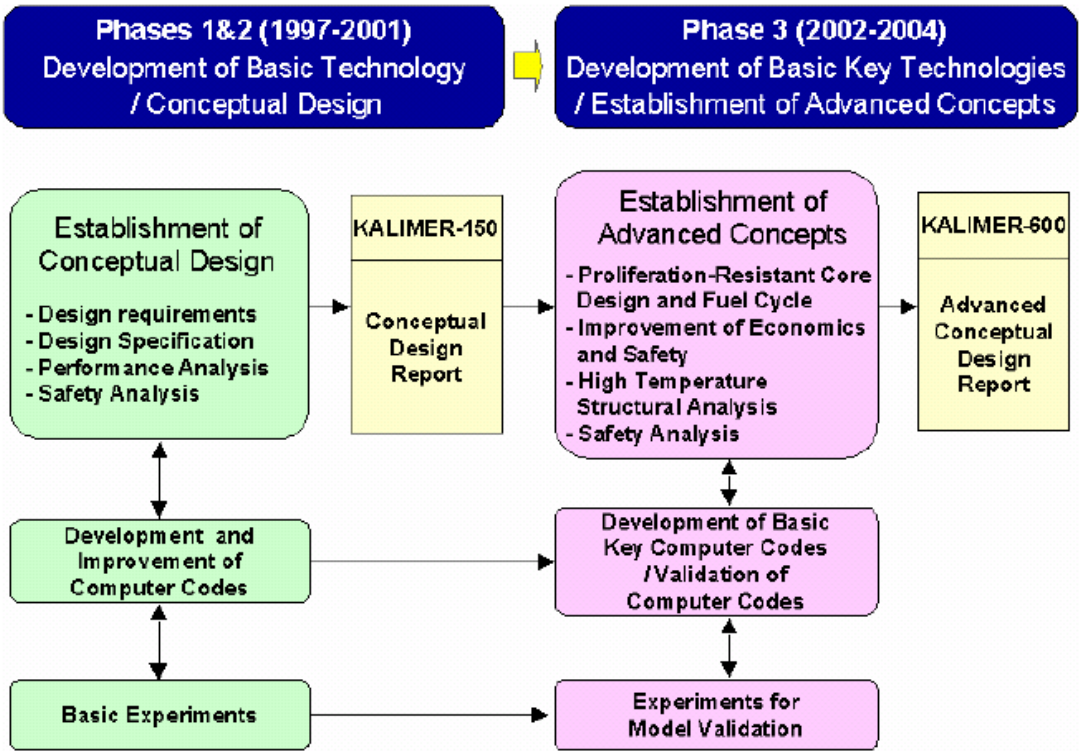
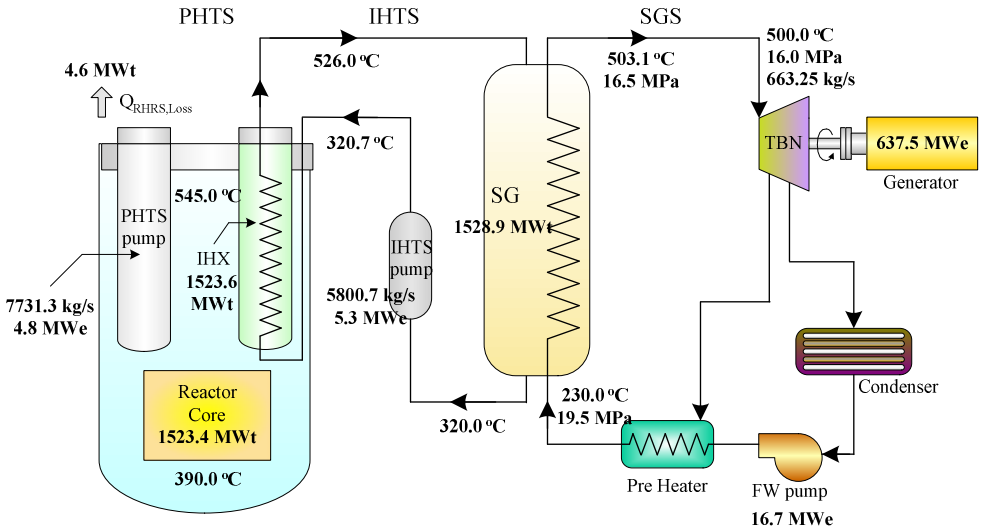


FIG. 2.46. Sodium cooled FR technology development in the Republic of Korea.

In Phase 3 from the year 2002 to 2004, basic key technologies and the advanced concepts of KALIMER-600 with a capacity of 600 MW(e) were developed (Fig. 2.47). During this phase, efforts were made to develop basic key technologies and to establish advanced concepts with emphasis on:

- proliferation resistant core design and fuel cycle concepts;
- system design for enhancement of economics and safety;
- high temperature structural analysis technologies;
- safety analysis methodologies.



- **Net Plant Power = 600.0 MWe**
- **Gross Efficiency = 41.9 %**
- **Net Efficiency = 39.4 %**

FIG. 2.47 Design concept of KALIMER-600 system.

In 2003, the preliminary KALIMER-600 design concept was established, and several experiments were performed for the validation of computer codes and models.

The 3rd national mid and long term nuclear R&D programme is currently being finalized in order to establish a detailed R&D plan for the years from 2007 to 2016. It is envisioned that the SFR technology development in the Republic of Korea will enter a new phase from 2007 with the participation in the Generation IV SFR collaboration.

Several components of KALIMER-600 have been designed in 2007 [35], e.g. the core with a single enrichment without blanket assemblies and a passive residual heat removal system. The inherent safety characteristics have been verified by a safety analysis.

Based upon the experiences gained during the development of the conceptual designs for KALIMER, KAERI is planning to develop advanced design concepts. There are three main categories of R&D activities under consideration:

1) conceptual design of an advanced SFR;
2) development of advanced SFR technologies necessary for its commercialization;
3) development of basic key technologies.

According to the current draft of the national mid and long term nuclear R&D programme, the objective of the SFR R&D programme is to develop an advanced SFR conceptual design with a schedule consistent with Generation IV SFR collaboration. The R&D endpoints, which are specified in the Generation IV Technology Roadmap Report, will be utilized to define the design activities and deliverables. The KALIMER-600 design will serve as a starting point for this effort, and a new advanced design will be developed to have features with a potential to better meet the Generation IV technology goals of sustainability, safety and reliability, economics and proliferation resistance and a physical protection.

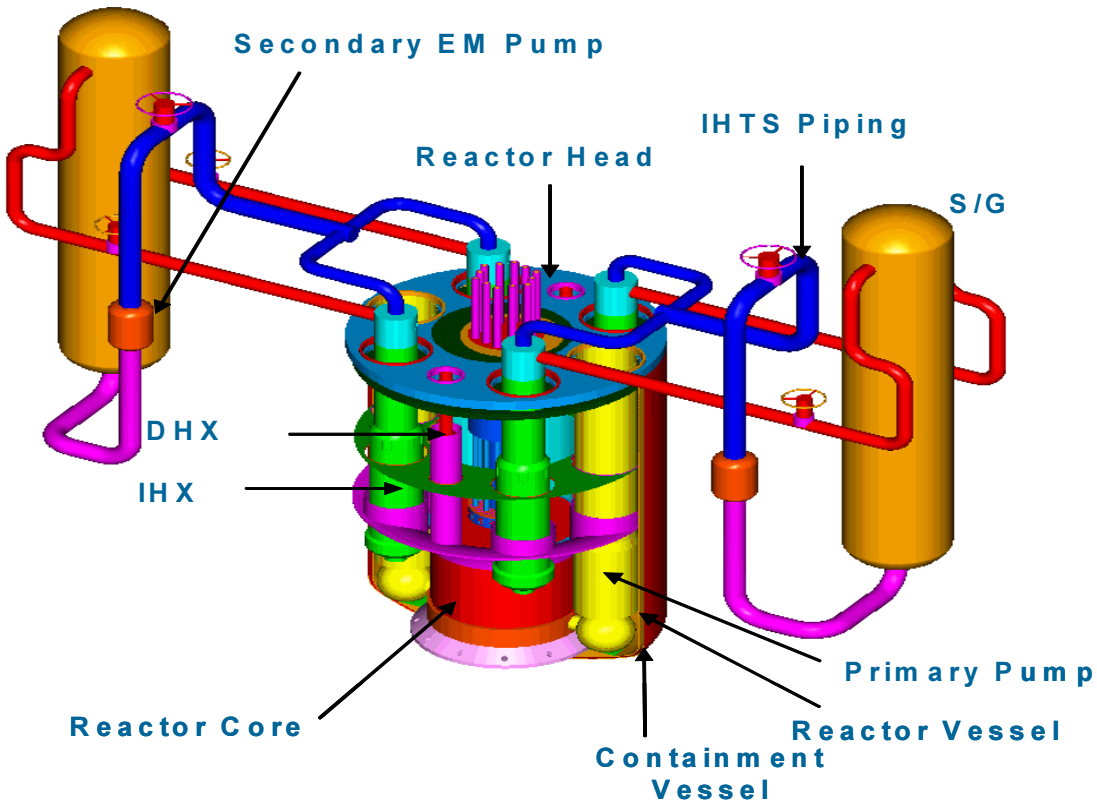


FIG. 2.48. Arrangement of KALIMER-600 systems and components.

For the development of advanced technologies, there are four areas: safety, fuels and materials, reactor systems and balance-of-plant. R&D will be performed for the improvement of economics and assurance of safety. For the development of the basic key technologies, the main focus will be on validating the computational tools and developing sodium technologies.

In addition to the KALIMER-600 a smaller version of a SFR is being studied with 150 MW(e) [11].

Lead cooled fast reactors

As an alternative fast spectrum reactor to the SFR, a project to study Pb alloy cooled fast reactors (called PEACER, Ref. [11]) was launched in 2002. This project aimed at developing the system design of the medium-sized integral power reactor that can produce electricity as well as transmuting transuranics and long lived fission products.

Preliminary designs on the 900 MW(th) Pb cooled breakeven and TRU transmutation core have been performed and further studies on alternative coolants, analysis of anticipated transients without scram events, basic cross-section verification and feasibility studies on passive decay heat removal are ongoing.

The preliminary neutronic designs of the 900 MW(th) Pb alloy cooled breakeven cores with nitride and metal fuel were performed. The cores were designed such that their conversion ratios (CR) with no blanket assemblies are slightly larger than unity to minimize the amount of excess fissile materials and fresh single enrichment fuels to be loaded. The flattening of power distributions is achieved by either using a fixed number of tie rods (TR) filled with B_4C burnable absorber (BA) or changing the number of empty tie rods (NST) within an assembly. In the aforementioned designs, the zone-wise conversion ratios are found to be slightly different. In a core with the single enrichment, CR values are desired to be constant throughout the whole core so that a discharged fuel might be reprocessed independently regardless of its origin. This goal was fulfilled by filling the space within the empty tie rods with graphite moderator. In this way, the zone-wise CRs are different at most by 0.006.

In addition to the breakeven cores, the design studies for a TRU transmutation core were performed in case of metallic fuel with different coolants of Pb and Pb–Mg. Analyses on the plant efficiency according to the operating temperature regime and a preliminary evaluation of reactor and guard vessel integrity upon the earthquake were done to derive the best possible combination of an operating temperature and a thickness of each component. With a modified version of SSC-K for the lead coolant, ATWS events analyzed showed no significant safety issues except that the passive decay heat removal using air cooling demanded a large vessel diameter as wide as 9 m. With the large volume of heavy metal coolant in the reactor vessel, a concern was raised on the sloshing of the reactor vessel upon an earthquake.

To investigate the corrosion behaviour of the (core) structure materials such as HT-9, T91 and 316L at high temperature, KAERI constructed three lead alloy corrosion test facilities. They are also used to develop methods to prevent corrosion if the corrosion damage is too severe. One method is to form a stable oxide layer on the material through oxygen control in the lead alloy. The first one is a static test facility, the second a Pb–Bi loop, and the third a Pb loop. The lead alloy loop is called KPAL (KAERI Pb Alloy Loop). The Pb–Bi and Pb loop are named KPAL-I and KPAL-II, respectively.

The static test facility KPAL is mainly composed of tube furnaces, a gas system and a glove box. The furnace is a 3 zone type. It minimizes the temperature variation (±1 K) for each section along the test tube made of quartz. The capacity is 10 kW and the maximum operation temperature is 800 C. There are two furnaces which are connected to the glove box. Oxygen

control is performed by flowing Ar/H_2 gas mixed with H_2O vapour. The static corrosion test facility also can be used as the facility to calibrate the oxygen sensor which is used to measure the oxygen content inside lead alloy.

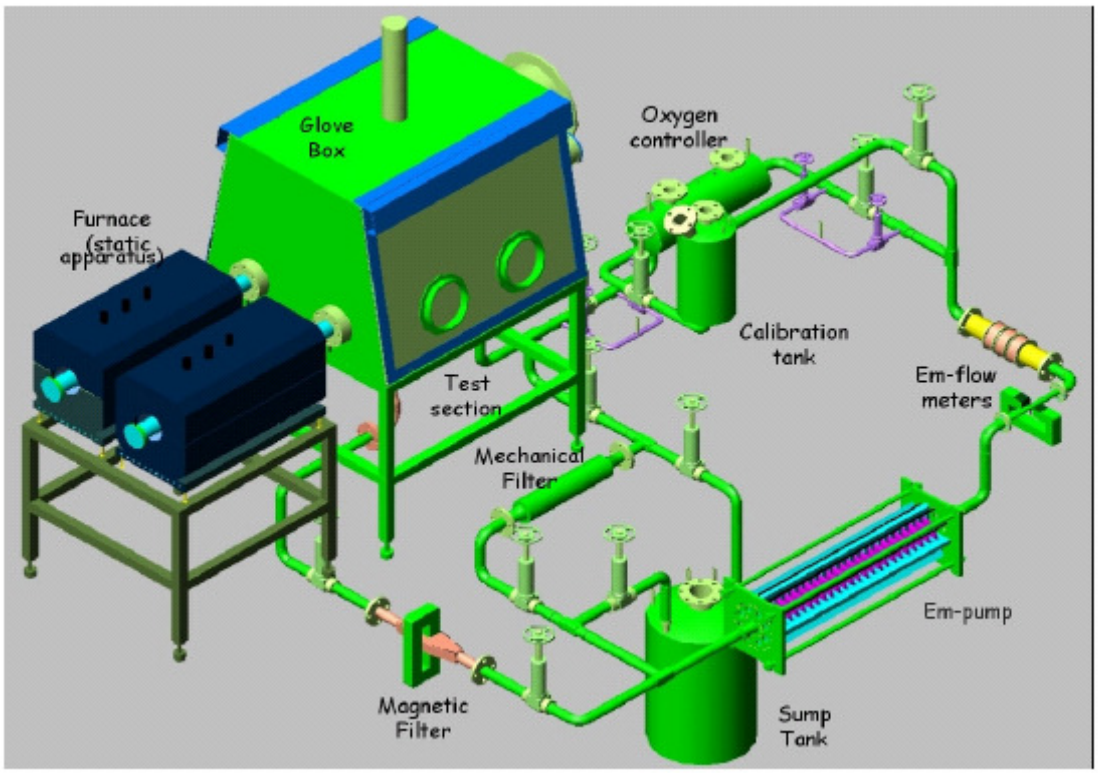


FIG. 2.49. KAERI's Pb–Bi corrosion loop KPAL-I.

The Pb–Bi loop KPAL-I (Fig. 2.49) is an isothermal loop. The flow velocity in the test section was designed to be around 2m/s in the range of 450~550°C and the charging volume of the Pb–Bi is around 0.03 m^3. The Pb–Bi loop is mainly composed of a main test-loop, bypass-loop for filtering Pb–Bi and a mixture-gas supplying system. The liquid metal in the main test loop circulates in the following order: Electromagnetic (EM) pump — EM flow meter — oxygen controller — test section — magnetic filter —EM pump. The Pb–Bi loop is being tested and the operation started in 2006.

The Pb loop KPAL-II is designed to be operated with flexible ΔT, ΔT_{hot} and ΔT_{cold}. The minimum ΔT_{cold} and maximum ΔT_{hot} are 450 C and 600 C, respectively. The Pb loop consists of an EMP, flow meter, heater, cooler and oxygen control system. EMP and EM flow meter will have the same configuration as the ones used for the Pb–Bi loop. The construction of the isothermal part of the Pb loop was completed and test runs were performed for 300 hours at 450°C. The mixed gas of 14% H_2O vapour and H_2 was forced to flow over the surface of Pb inside the oxygen control tank. The oxygen content of the Pb was estimated to be 4.0×10^{-7}wt%. The non-isothermal part of the Pb loop was completed in 2006.

2.5.7. FR development in the Russian Federation

The Russian Federation's nuclear energy policy, carried out by Rosatom, is defined in the programme for nuclear power development in the Russian Federation in 1998–2005 and in the period up to the year 2010. The programme sets forth the objectives of assuring safe and cost-effective operation of the nuclear power complex and developing advanced NPPs for construction in the next decade.

Figure 2.50 shows a tentative scenario of the growth of nuclear power in the Russian Federation including deployment of fast reactors.

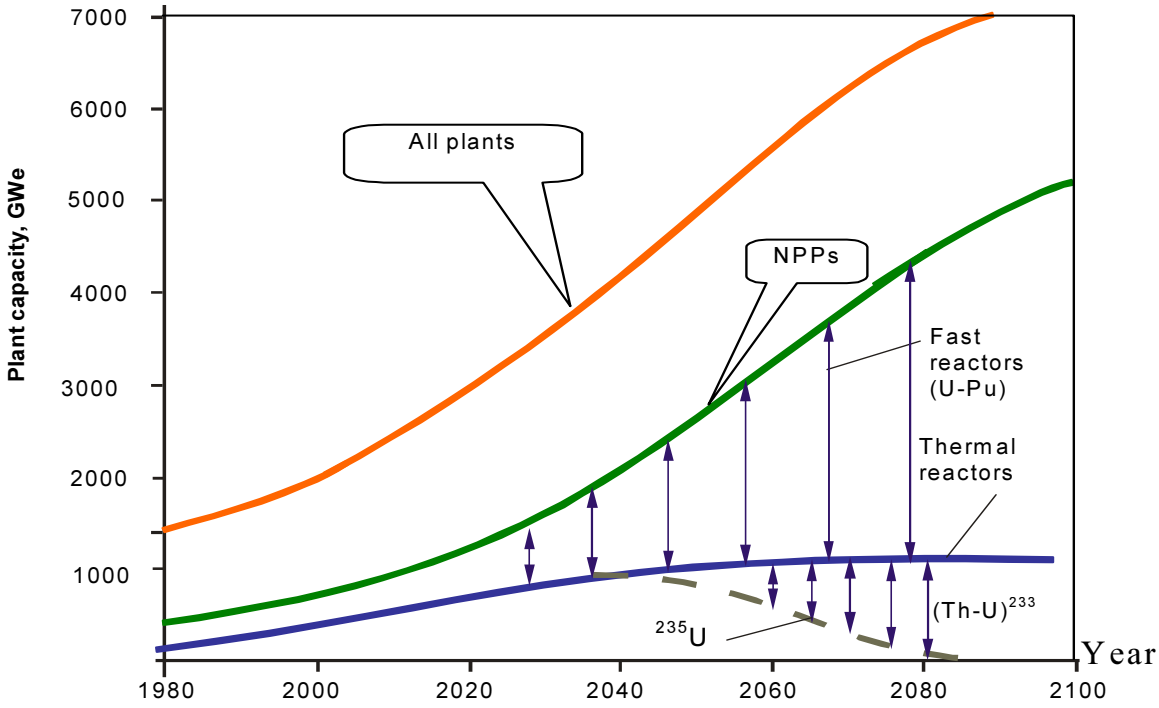


FIG. 2.50. Tentative scenario of nuclear capacity growth, including fast reactors.

Currently, there are two fast reactors in operation in the Russian Federation (BOR-60 reactor in Dimitrovgrad and BN-600 reactor in Zarechny). The first fast reactor in the Russian Federation was the BR-10.

BR-10 experimental reactor

The experimental BR-5 reactor with a thermal power of 5 MW was put into operation in IPPE in 1959. After modernization in 1971–1973, the reactor was renamed BR-10 and its power had increased up to 8 MW. After operation during about 44 years BR-10 was finally shut down on 6 December 2002 and is currently prepared for its decommissioning.

Current status of the BR-10 reactor is as follows:

- all fuel subassemblies have been unloaded from the core and replaced by the dummy subassemblies;
- sodium has been drained from the primary and secondary circuits to the storage tanks and frozen;
- the inner surfaces of the primary circuit have been cleaned from sodium and decontaminated.

Figure 2.51 shows the history of development of Russian fast reactors after the BR-10.

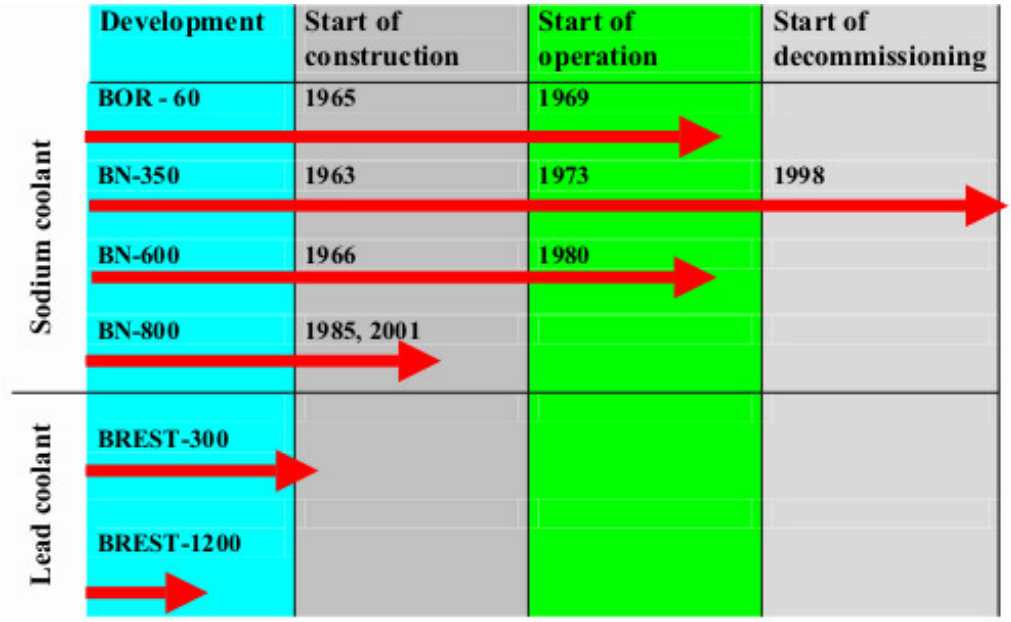


FIG. 2.51. Development of fast reactors in the Russian Federation.

BOR-60 experimental reactor

The BOR-60 reactor started up in 1968. In 2003, BOR-60 reactor power was up to 55 MW. According to the decision made by the government, BOR-60 reactor operation is permitted till 31 December 2009, taking into account measures on assurance of safe operation.

In the framework of preparation for BOR-60 licence prolongation and substantiation of its safety the following activities were performed in 2005:

- creation of an updated and approved list of initial events of beyond design basis accidents that should be analyzed for BOR-60 safety justification;
- analysis of an external shock event of rated magnitude (10 kPa) as applied to the reactor building and preliminary estimation of response spectra and acceleration;
- irradiation of samples of BOR-60 reactor vessel material;
- design and fabrication of packages for removal of cesium radionuclides from the primary coolant, and testing of the procedure of coolant purification;
- inspection of condition of various components and systems and extension of their lifetime, including:
  - performance of technical certification of CSS elements and equipment of BOR-60 reactor;
  - inspection of welds in the secondary loops and replacement of elbows expansion joint on the secondary pipeline.

After 2009, two stages of BOR-60 reactor adaptation are planned in order to replace it with the new reactor BOR-60M.

On the first stage, there will be special activities for startup including:

- BOR-60 reactor decommissioning (putting in final storage);
- construction of BOR-60M reactor with all systems assuring its design operating conditions and connection to the existing primary circuit;
- creation of additional systems and bringing of existing systems into the accordance with requirements of regulatory documents.

The 1st stage includes complete replacement of safety-related components and systems (primary and secondary loops, fuel handling system, auxiliary systems, measuring equipment, power supply systems, automatic control systems etc.).

The 2nd stage implies adaptation of water-steam circuit and systems not influencing safety, which will be converted as their lifetime is expired. During 2003, BOR-60M basic design was developed.

Figure 2.51 provides an overview of the planned development of fast BN reactors.

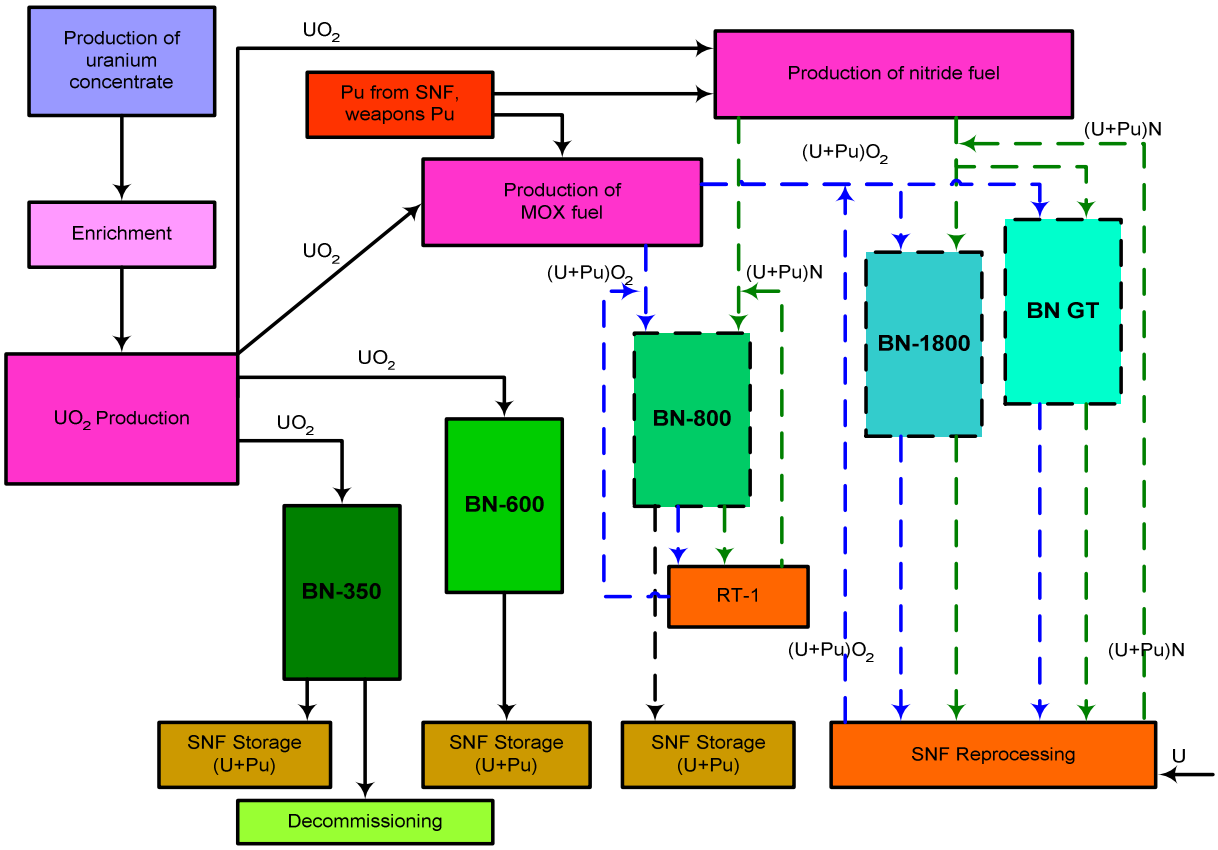


FIG. 2.51. Scheme of BN technology development in the Russian Federation.

BN-350 reactor decommissioning

The BN-350 reactor was in operation from 1972 till 1999, supplying heat, electricity and water to a nearby city of Aktau. Currently, Russian institutions participating in the work on designing, construction and supervision of operation of the BN-350 reactor (located in Kazakhstan) are invited for consultations and participation in the work on the BN-350 reactor decommissioning.

Reactor BN-600

The BN-600 reactor started up in 1980. Design lifetime of the BN-600 reactor plant (Fig. 2.52) expires in April 2010. During the period 2002–2005, the first stage of the BN-600 lifetime extension (LTE) by 15 years was accomplished.

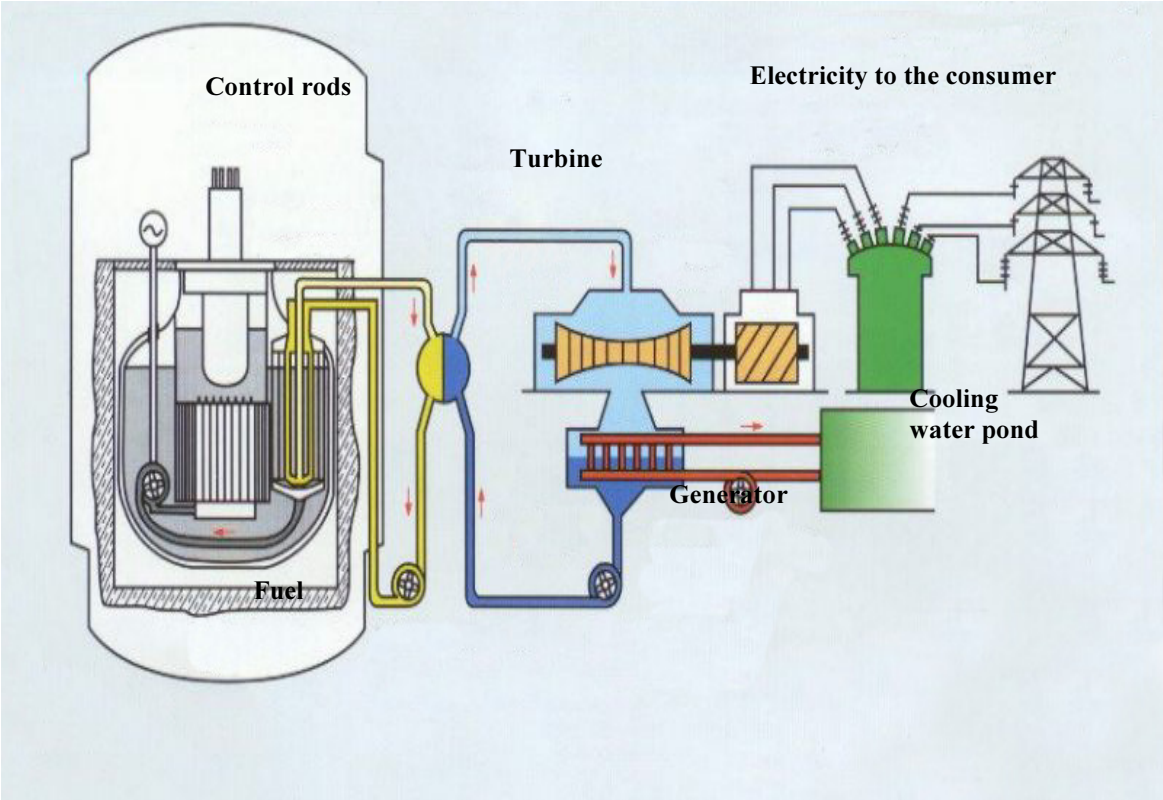


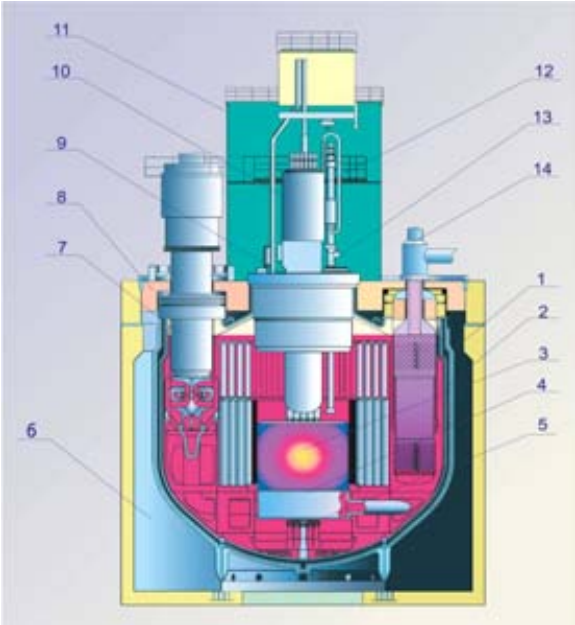
FIG. 2.52. BN-600 nuclear power reactor.

Results of performed work (comprehensive examination of the systems and components of the power unit and estimation of its current safety level) were used to form in 2005 the basis of the investment project on LTE showing cost effectiveness and attractiveness of further work in this area. According to the estimates, specific cost of LTE of the BN-600 power unit is ~US $250 per kW of installed power. The Russian Federation plans to reconfigure the BN-600 to burn weapon-plutonium.

BN-800 reactor

Construction of BN-800 (Fig. 2.53) as unit number 4 at Beloyarsk site is currently underway. It is for the first time in practice of the contemporary Russian Federation that construction of this unit is under the intense attention of the leadership of the country. Moreover, the guaranteed financial support of the BN-800 construction is from the state budget with increasing rates with time. Commission of the BN-800 is due in 2012. With commissioning of the BN-800 reactor four directions can be identified providing additional impulse of development and perfection of FR:

- increasing safety;
- improving economic parameters of fast reactors and their nuclear fuel cycle;
- mastering of MOX fuel fabrication at industrial scale;
- mastering of closed fuel cycle elements for innovative nuclear power systems.



1. Vessel
2. Guard vessel
3. Reactor core
4. Pressure chamber
5. Confinement device
6. Vault
7. Main circulation pump
8. Upper fixed shielding
9. Large rotation plug
10. Central rotation plug
11. Protection hood
12. Reloading mechanism
13. Small rotation plug
14. Intermediate heat exchanger

FIG. 2.53. The reactor BN-800.

The BN-800 design uses all the principal technical solutions that have been realized in the BN-600 reactor and proven by its operation during many years. At the same time there are new technical solutions envisaged in the BN-800 reactor which substantially increase nuclear safety. Such innovative solutions are the following:

- The first time introduction by design of a zero or even negative figure of the sodium void reactivity effect by changing size and structure of the upper axial blanket so that in case of sodium boiling in the fuel subassembly, the neutron leakage sharply increases upward and out of the core.
- Addition of scram rods based on passive activation into the reactor control system so that once forced coolant circulation stops these scram rods fall down into the core under gravity and shut down the reactor.

A proposal was considered giving international status to the BN-800 reactor project by offering interested countries to join within a framework of international cooperation.

BN-1800

The Russian Federation also plans to develop BN-1800 (Fig. 2.54) in order to:

- maximize the use of gained experience (pool type reactor design, three-circuit heat removal system, design of the main components);
- increase of unit power up to 1800 MW(e) (taking into account trends in the development of thermal reactor design etc.);
- increase of thermal efficiency up to ~46%;
- decrease of the number of systems and components;aiming at high efficiency fuel cycle using high density fuel.

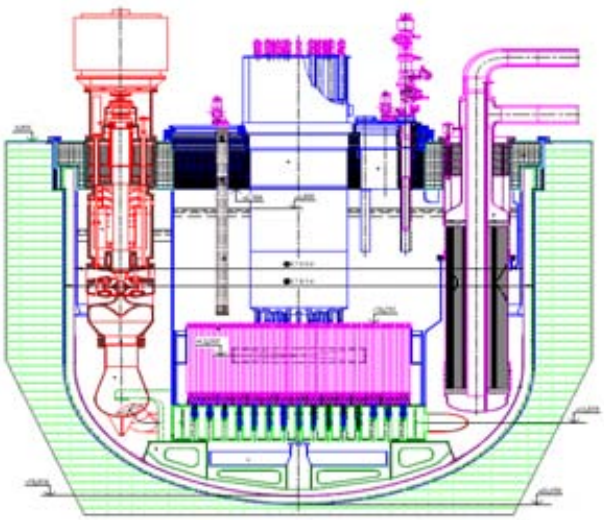


FIG. 2.54 Advanced sodium cooled fast reactor BN-1800.

Lead cooled BREST-OD-300 reactor

The Russian Federation has experimented with various lead cooled FR cores and has a lot of experience with lead–bismuth cooling from 41 years of operating submarine reactors.

The basic specific features of the BREST fast reactors are:

- use of high density and high thermal conductivity mono-nitride fuel;
- lack of uranium blanket;
- use of lead coolant and reflector;
- uranium enrichment is not required.

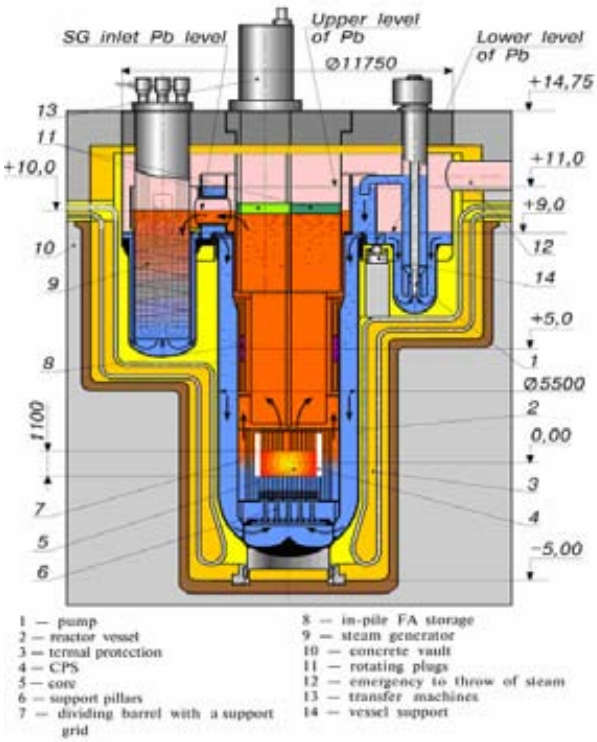


FIG. 2.55. BREST-300 reactor.

It has a design power of 300 MW(e) or more (Fig. 2.55), uses supercritical steam generators and (U + Pu)-nitride fuel. It is inherently safe; no weapon grade Pu can be produced (there is no U-blanket) and spent fuel can be recycled indefinitely.

By today, the conceptual designs have been prepared for the 300 MW(e) and 1200 MW(e) modifications of the BREST reactor, design and calculation studies have been performed. Reactor physics was validated in experiments at U–Pu–Pb critical test facilities, and nuclear data were corrected. Long term corrosion testing of steels was performed in circulation Pb loops and experiments were carried out to study the interaction of Pb with air and water, as well as nitride fuel with Pb and steel cladding, etc. A pilot plant is planned to be built at Beloyarsk and a 1200 MW(e) unit is designed.

SVBR-75/100 reactor

Within the framework of conceptual design of a multi-module NPP with two power units of 1600 MW(e) each, development of the conceptual design of SVBR-75/100 reactor with lead–bismuth as coolant was completed. Currently, the concept of a seaside nuclear desalination power complex is under development on the basis of the SVBR-75/100 reactor with scientific supervision by the SSC RF-IPPE (Fig. 2.56). Preliminary estimates have shown that construction of such complexes can be profitable for developing countries lacking electricity and fresh water.

It is envisaged within this concept that desalination and turbogenerator units are constructed at the expense of country user (customer) and connected to a transportable reactor unit (TRU) manufactured in the Russian Federation, equipped and transported with frozen coolant to the site and provided to the country for operation on a leasehold basis. Upon completion of its lifetime, TRU is replaced with a new one and returned to the Russian Federation for refuelling, preventive repair work and preparation for the next operation cycle.

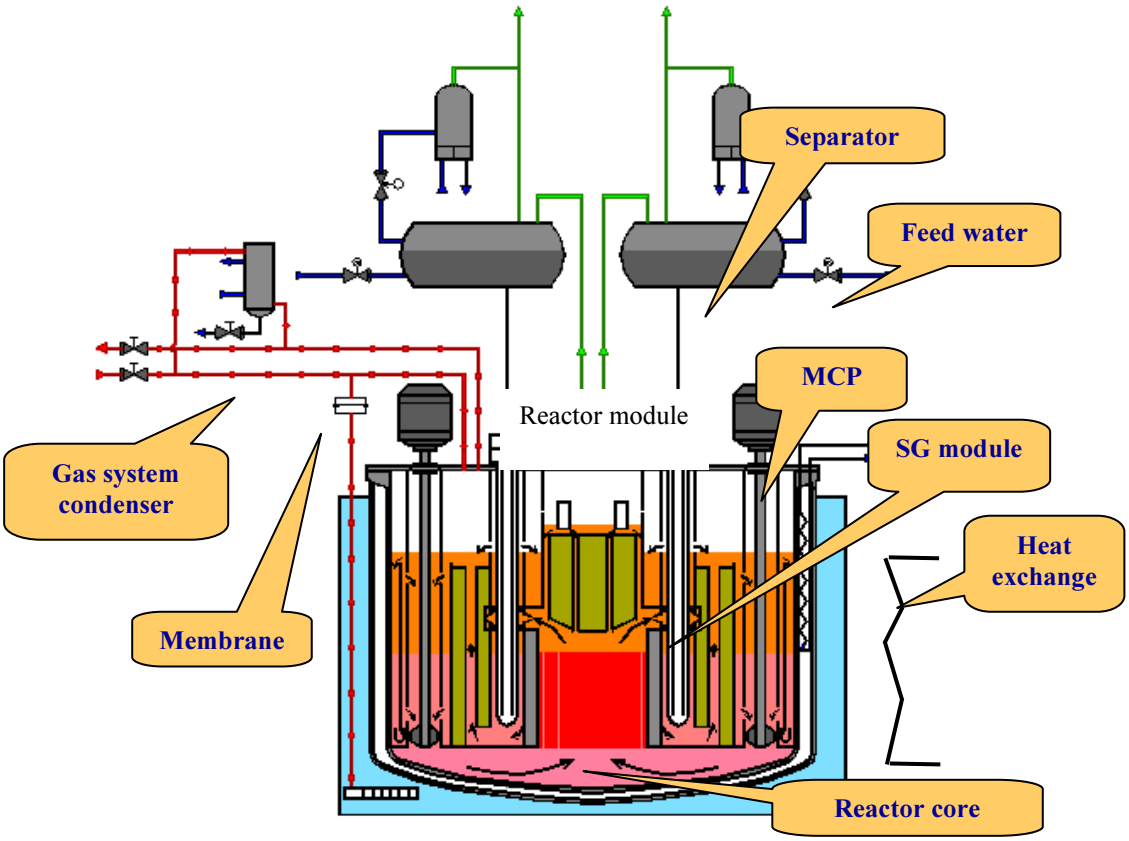


FIG. 2.56. SVBR-75/100 RI flow diagram.

Two-circuit NPP with sodium cooled fast reactor and gas turbine

One of possible ways of improvement of sodium cooled fast reactor technology and extension of area of their application is the use of a two-circuit heat removal system design (instead of three-circuit design) with a gas turbine. This would make it possible:

- To improve reactor safety by eliminating the risk of sodium–water contact.
- To simplify power unit design, decrease specific steel consumption owing to elimination of an intermediate circuit with steam generators and relevant reduction of the number of safety systems and, hence, significantly decrease capital cost.
- To increase efficiency of power unit by increasing upper temperature level, using a gas-turbine cycle instead of a steam-water cycle and upgrading the thermodynamic cycle of the plant by approaching an ideal cycle (Carnot cycle).
- To expand the area of fast reactor application by means of increasing parameters of primary sodium and production of high temperature process heat usable for various technologies, such as that of hydrogen production.

Conceptual design studies on a two-circuit NPP with sodium cooled fast reactor with a gas-turbine cycle were carried out at the SSC RF-IPPE in two areas: large size NPP (1200–1600 MW(e)); small and medium size modular transportable cogeneration NPP (up to 300 MW(e) (ATES-BN-GT-300/100) (Fig. 2.57).

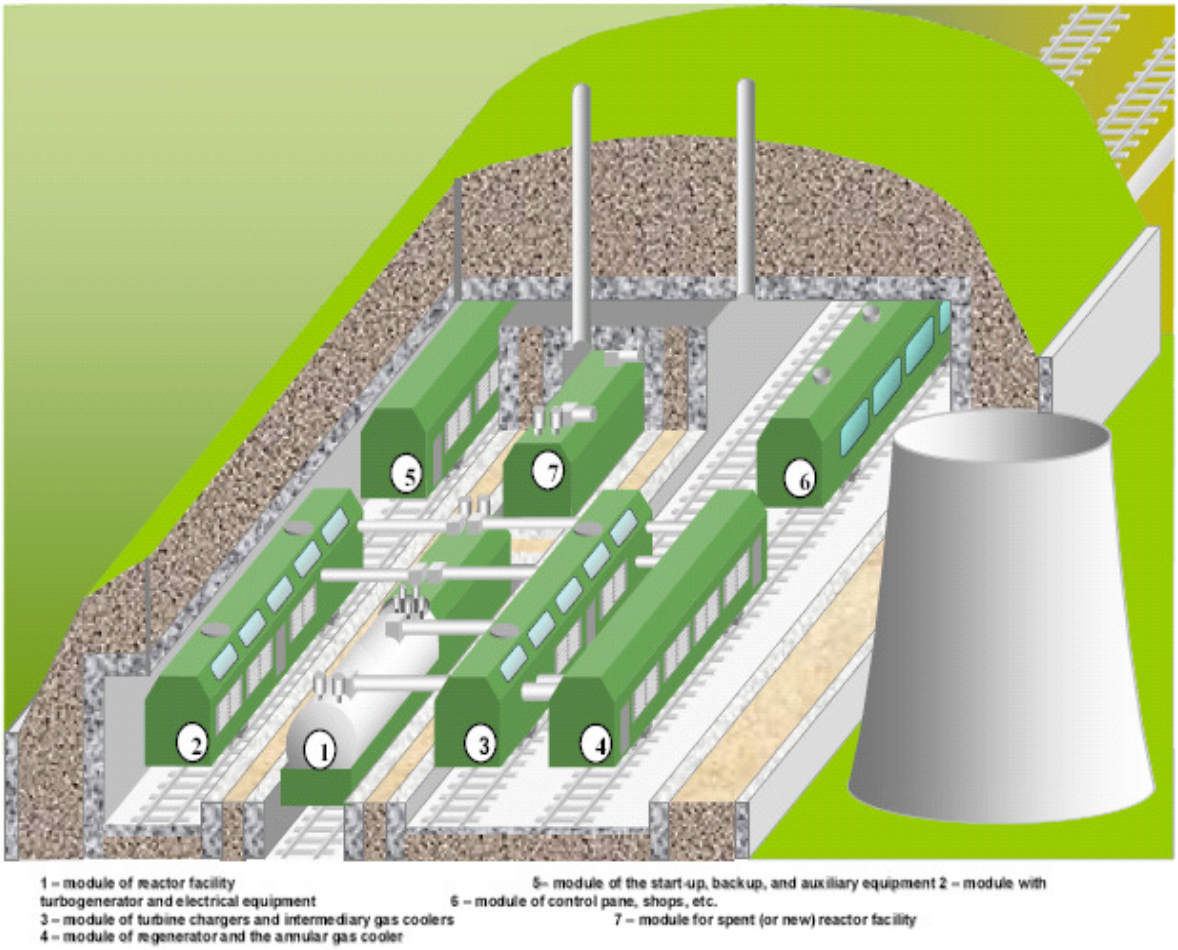


FIG. 2.57. NPP with BNGT-300/100.

Change-over to a two-circuit layout with a gas-turbine cycle will provide, on the one hand, complete implementation of sodium coolant benefits for cooling the reactor core with a

fast-neutron spectrum at a high enough temperature level; on the other hand, it will minimize sodium coolant drawbacks due to its chemical reactivity against air and water (i.e. primary circuit should be completely integrated; heat removal system should operate without water inventory).

WWER-SCWR fast reactor

Concept of a once-through reactor cooled with supercritical water (SCWR) is promising because of its following advantages:

- high efficiency of power unit;
- low specific steel consumption of reactor plant using one-circuit NPP arrangement with vessel-type reactor design;
- high indices of fuel use owing to the application of supercritical pressure and tight fuel pin lattice.

Conceptual analysis of a reactor design based on the use supper critical water as a coolant (called WWER-SCP) has been performed at the SSC RF–IPPE. The main features of this design are as follows:

- one-circuit NPP arrangement using once-through coolant flow;
- two turbines with steam re-heater;
- vessel-type reactor design;
- tight lattice of MOX fuel pins assuring sufficiently high energy neutron spectrum (close to that of fast reactor) and fuel breeding ratio close to one (0.94);
- presence of internal blankets of depleted uranium and zirconium hydride as interior axial layers in the core.

2.5.8. FR development in the UK

The UK continues to support the international development of fast reactor technology, mainly through participation in European and international collaborations. All work is currently funded by BNFL, and is carried out by BNFL, NNC, and SERCO Assurance (formerly AEA-Technology). UKAEA provides assistance with archiving activities related to the originally government-funded fast reactor development programme, which was terminated in 1994.

The principal programmes of work related to fast reactor and accelerator-driven systems are as follows.

The CEA-led CAPRA programme initially focused on the incineration of Pu in a fast reactor core. Subsequently, the complimentary CADRA programme was initiated, with the aim of broadening the scope of CAPRA to include the incineration of minor actinides and long lived fission products. UK participation in these programmes has been focused in the areas of core physics, fuel performance modelling, and fuel cycle (scenario) modelling.

UK organizations participate in several EURATOM 5th Framework programmes which are concerned with fast reactors or accelerator-driven systems, and their associated fuel cycle technologies. Examples of these include:

- GCFR — gas cooled fast reactor study (completed in 2003);
- MUSE — basic nuclear data measurements in the CEA Masurca facility;
- CONFIRM — fabrication and in-reactor performance of Pu and minor-actinide nitride fuels;
- FUTURE — studies of fabrication and properties of transuranic oxide fuels;

- PDS-XADS — accelerator-driven systems technology and design studies.

In addition, BNFL and its partners have participated in the preparation of several proposals for the 6th Framework programme, and already participate in the RedACT study, which examines the technical options and consequences of alternative strategies for the management of actinides, including the use of fast transmuter systems.

The UK participated in the development of the International Generation IV Technology Roadmap, and will participate in the R&D programmes which are currently under discussion amongst the GIF partners. Amongst the 6 concepts identified for development under Generation IV, the UK has indicated that it will participate in the following reactor development programmes:

- the very high temperature reactor, VHTR (seen as a natural continuation of the UK's interests in graphite moderated gas cooled systems, and complimentary to nearer-term interests in the PBMR);
- the gas cooled fast reactor, GFR;
- the sodium cooled fast reactor, SFR.

Currently, R&D plans for each of these systems are nearing completion, and plans for conducting the R&D collaborations are under consideration. Task sheets have been prepared and submitted to the coordinators for each system, detailing the proposed R&D contributions in each area (design and safety, fuel, materials, fuel cycle, balance of plant, etc.).

In addition to these multilateral activities, BNFL pursues bilateral collaborations with CEA and JNC. These are wide in scope, but include aspects of fast reactor technology. It is likely that in the future, the UK will conduct the majority of its research into fast neutron systems within the Euratom Framework Programme and within the Generation IV collaboration.

The Dounreay site was opened in 1955 in support of the UK's fast reactor programme, and three reactors were built over the next 20 years — the Dounreay fast reactor (DFR), prototype fast reactor (PFR) and the Dounreay materials test reactor (DMTR). All are now closed, and management at the site is now focused on decommissioning the reactors, ancillary nuclear facilities and the restoration of the environment. This represents a major activity, and staffing levels have risen from 1100 in the mid-1990s to over 2000 at present. The Dounreay Site Restoration Plan is expected to take 50–60 years to complete and cost in the region of £4 billion.

Construction of the sodium disposal plant (SDP) has been completed at a cost of some £17 million. The plant, which is situated in the former turbine hall of PFR, destroyed around 280 t of sodium during commissioning, and has now received regulator approval to start work on the 1500 t of sodium in PFR.

A water vapour nitrogen (WVN) process has been developed to allow the removal of pockets of sodium (Fig. 2.58) which will remain in the PFR vessel once sodium draining has been completed, and the process has been qualified during the decommissioning of the PFR sodium tank farm and secondary sodium circuits. Completion of the sodium removal and vessel cleaning is expected around 2009.

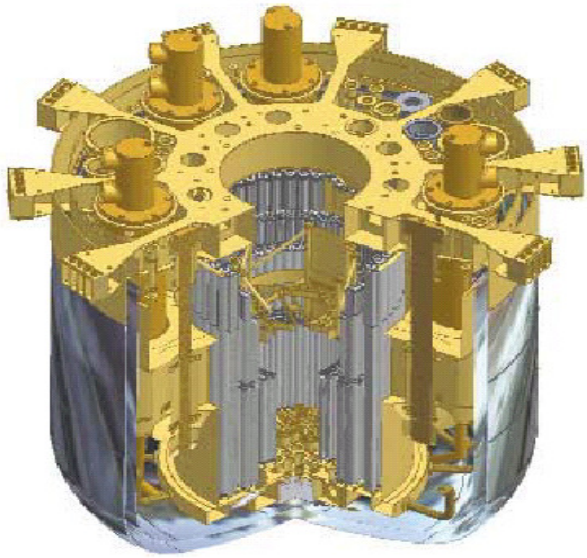


FIG. 2.58. Study of likely areas for remaining sodium pools in the PFR main vessel.

Decommissioning and demolition of the facility D1249 has now been completed. This facility was originally constructed to house critical assembly tests, and was subsequently modified to accommodate the Supernoah rig for the study of sodium–water reactions. The facility was last used for experiments to support the design of the EFR steam generators, which were completed in 1995.

2.5.9. FR development in the USA

Historically, there was an intensive development and deployment programme for FRs going on in the USA till the beginning of the 1990s. In total, five different FRs (see Table 2.11) have operated in the USA and additionally several more have been designed during that period. However, in the middle of the 1990s the operation of all FRs had been stopped, and only conceptual design studies (e.g. integral fast reactor) and development of fuel for FRs (AFCI, see Section 3.4.4) had been continued.

Currently, the USA is showing interest in pursuing FR again, as illustrated by two international activities initiated by the USA called Generation IV International Forum (GIF) and Global Nuclear Energy Partnership (GNEP), discussed further in Sections 2.6.1 and 2.6.2, respectively.

A primary goal of the current US national energy policy is to add supply from diverse sources (Fig. 2.59).

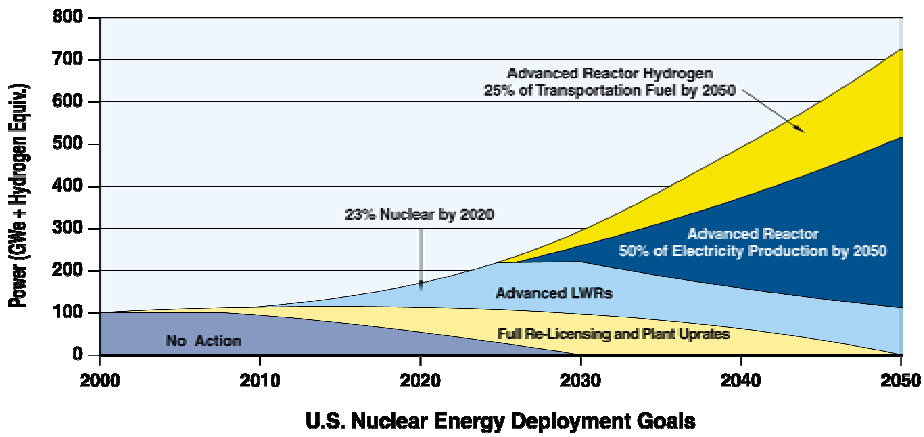


FIG. 2.59. US nuclear energy development goals.

It can be expected that the main stream of developing innovative nuclear energy systems in the USA will follow the Generation IV roadmap. The long term strategy of the USA in regard to nuclear energy is shown in Fig. 2.60.

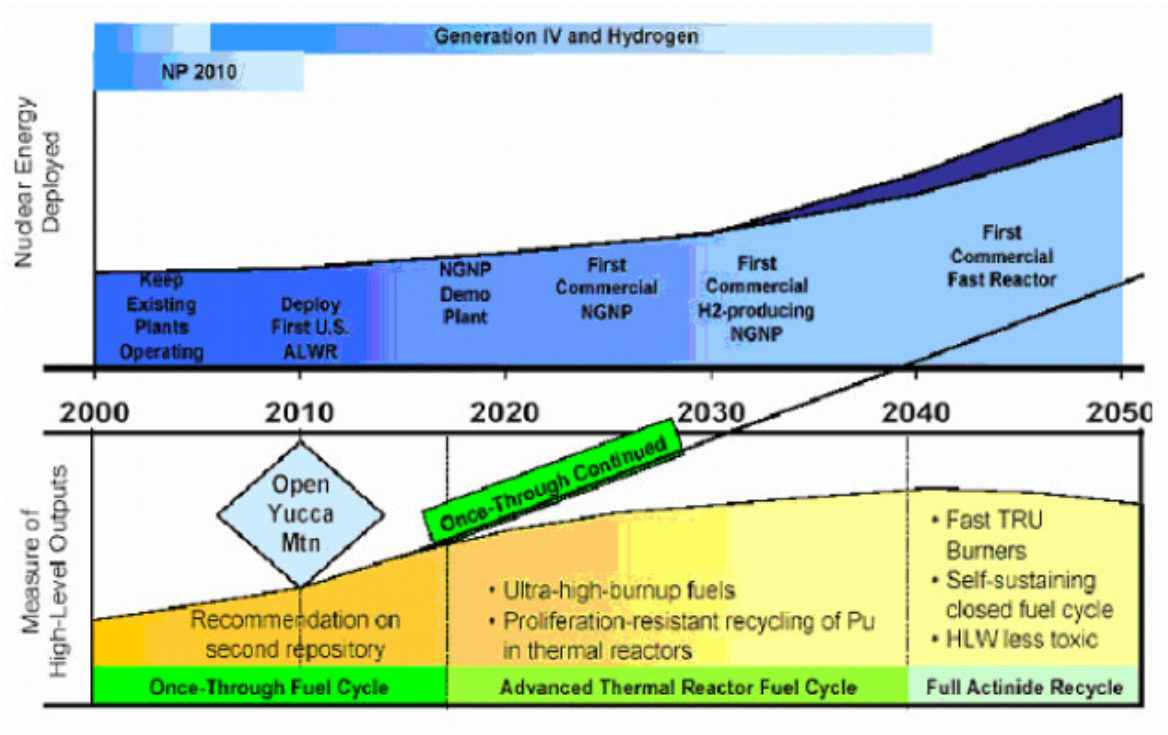


FIG. 2.60. Long term US strategy for nuclear energy.

The priority on fast reactors (Fig. 2.61) responds to the sustainability goals of the Generation IV Roadmap (to be discussed in more detail in Section 2.6.1), and reflects their excellent potential to make significant gains in reducing the volume and radiotoxicity, and increasing the manageability of spent nuclear fuel. With a successful fast reactor programme, the USA may be able to avoid the need for a second geological repository for many decades.

The most promising fast-spectrum Generation IV Systems are the gas fast reactor (GFR), the lead fast reactor (LFR) and the sodium fast reactor (SFR). Among these, the LFR and GFR are given the most emphasis in order to resolve technical issues and uncertainties, since these reactors offer strong potential benefits that have not been fully demonstrated.

The SFR is already at a fairly advanced state of development, with many of its technologies having been demonstrated internationally. All of these systems should be brought to a state where a down-selection can be undertaken based on demonstrated performance of their economics, safety and reliability, sustainability, and proliferation resistance and physical protection.

The remaining two technologies identified during the roadmap, the supercritical water cooled reactor (SCWR) and the molten salt reactor (MSR), are also included in the Generation IV USA programme. The current programme includes significant international collaborative efforts on the SCWR, and exploratory collaborations on the MSR.

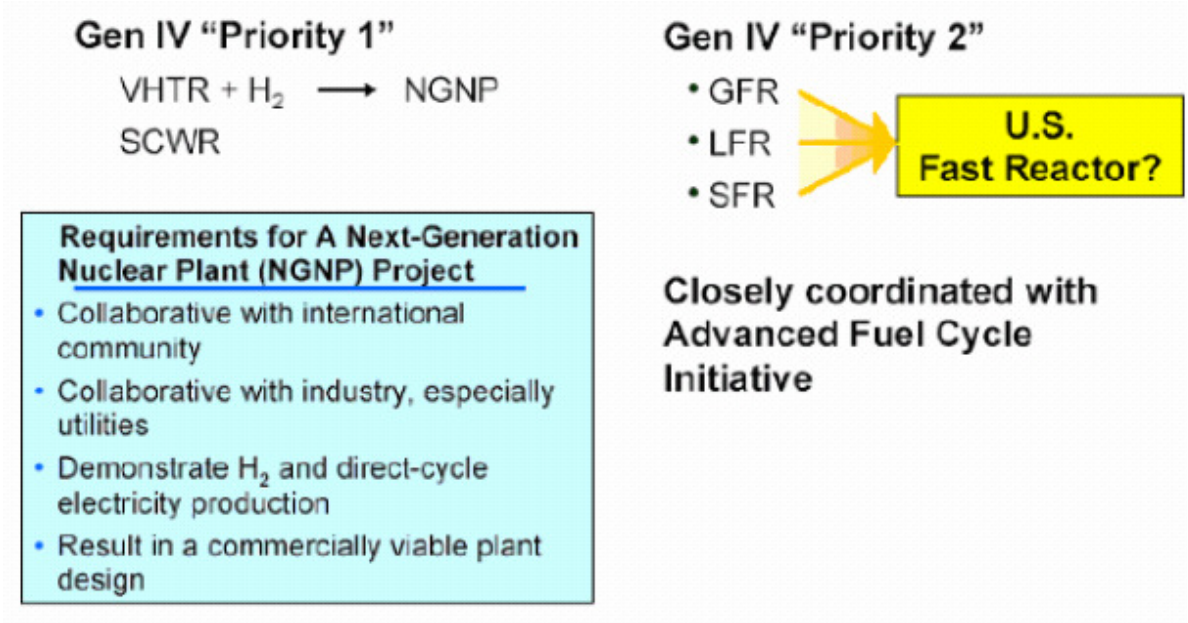


Figure 2.61. US DOE Generation IV priorities.

The Advanced Fuel Cycle Initiative (AFCI) is discussed further in Section 2.6.2.

Another example of a concept pursued in the USA is the Encapsulated Nuclear Heat Source (ENHS) (Fig. 2.62), as proposed by the Argon National laboratory (ANL) [36]. The idea is to provide a small rating, turnkey power plant with a liquid metal cooled fast reactor, which like a nuclear battery could be delivered already fuelled to the client's site, would provide energy for 15 to 20 years without refuelling.

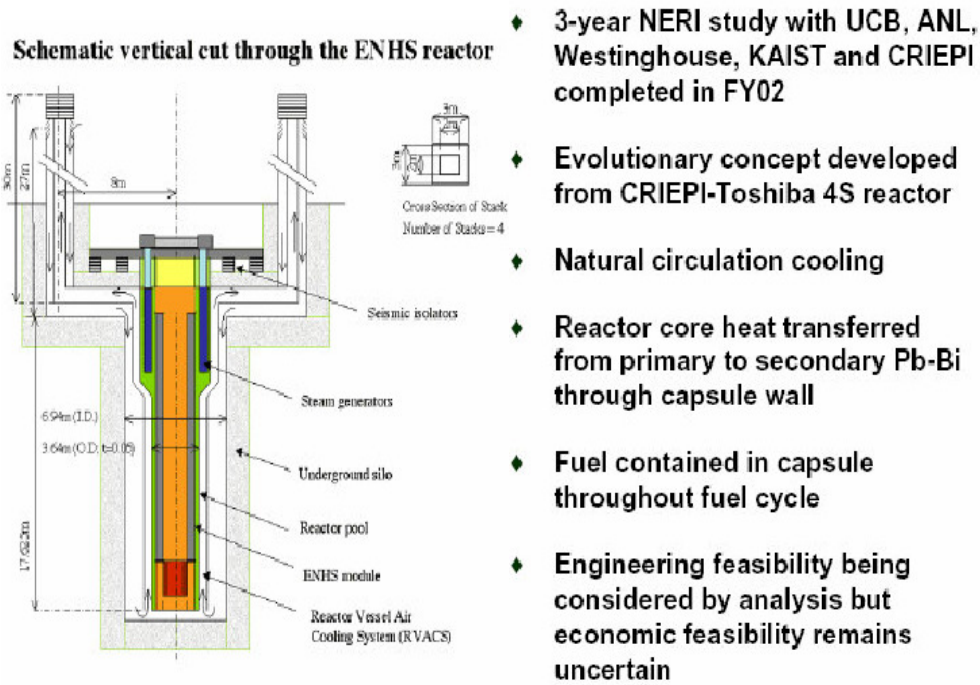


Figure 2.62. ENHS reactor.

At the end of lifetime it would be changed out for a replacement battery ENHS module and returned to a regional centre for back end fuel cycle services. The ENHS module would be inserted into a permanently sited secondary heat transport circuit driving a Rankine steam cycle at the client's site.

Figure 2.63 shows the concept of a transportable Star-LM HLMC Natural Convection Reactor, as proposed by the Argonne National Laboratory (ANL) [37].

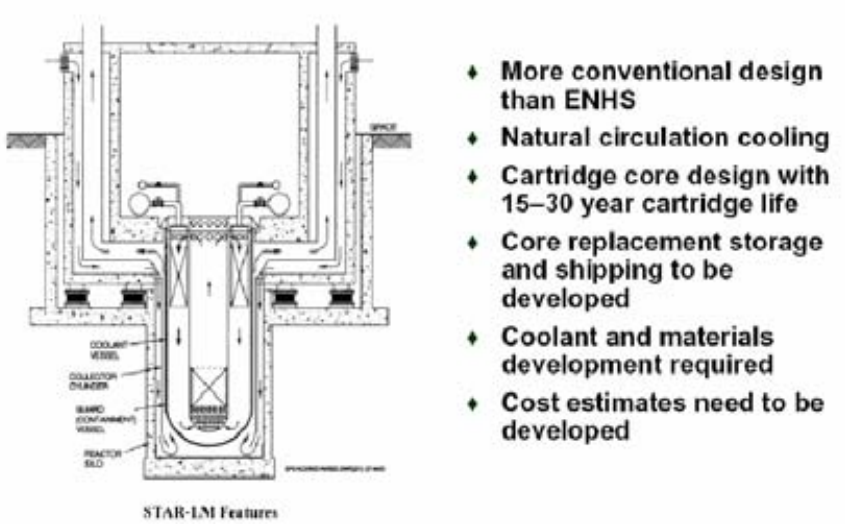


FIG. 2.63. STAR-LM features.

As of 2007, this concept is being further developed and renamed SSTAR with a power of 20 MW(e), using Pb as primary coolant and a supercritical S–CO_2 Brayton cycle energy conversion.

There was a Super PRISM[6] FBR under development by GE for a modular pool-type reactor with passive cooling. The pool-type module is cooled by sodium. The commercial concept used six reactor modules to provide 2280 MW(e). The design meets Generation IV requirements.

2.6. Multinational programmes for development of advanced reactor concepts

There are many considerations and studies going on internationally to prepare options and plans for the further development of nuclear energy technology in the near (15–25 years) and in the more distant (>25 years) future. Most of them are aware that this development has to take care of and to synthesize demands of sustainability, i.e. of economics, safety and security, environment, waste, infrastructure and proliferation resistance.

And most of them understand those efforts as part of necessary activities that include the whole nuclear fuel cycle. In this section, only the considerations regarding reactor systems are very shortly reviewed.

By far, the largest and strongest effort of this kind is the Generation IV and the GNEP initiative. However, there are also studies and investigations underway performed under the umbrella of EURATOM, bilateral Russian–US programmes and various national activities, as in France (e.g. gas cooled fast reactors) and in Japan.

[6] The name comes from a former modular 150 MW(e) liquid metal cooled inherently safe reactor (GE design). But this design and another Advanced liquid meal FBR (ALMR) of 1400 MW(e) (ANL + GE design) were withdrawn from NRC review at an early stage.

2.6.1. The Generation IV initiative

In 2001, an initiative has been started by the US Department of Energy (DOE) and the two US national laboratories Idaho National Engineering and Environmental Laboratory (INEEL) and Argonne National Laboratory (ANL) to promote an international cooperation in research for a future generation of nuclear energy systems, called Generation IV [38]. Figure 2.64 shows the basic idea of Generation IV as a new step after three generations of reactor technology in the past.

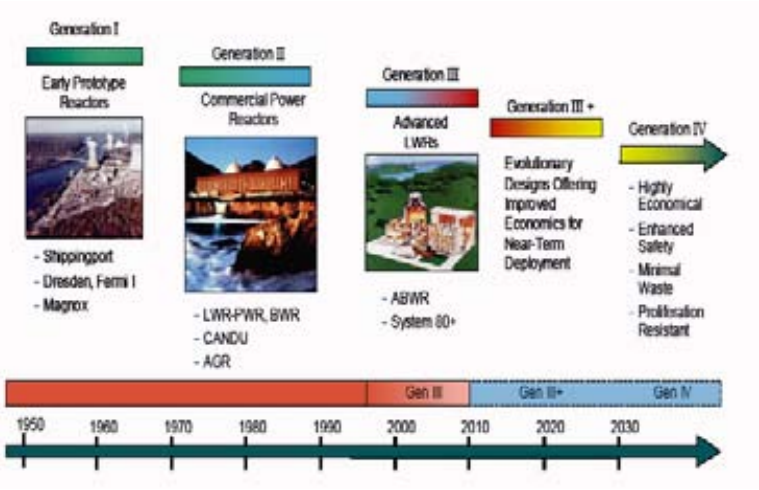


FIG. 2.64. Basic idea of Generation IV. A technology roadmap for Generation IV nuclear energy systems (December 2002).

The United States of America and seven other governments[7] of nations signed a charter creating the Generation IV International Forum (GIF) [39]. The charter establishes the GIF as an international group dedicated to developing the next generation of nuclear reactor and fuel cycle technologies over the next several decades.

Working Groups have been set up and organized as shown in Fig. 2.65[8].

[7] GIF starting members were: Argentina, Brazil, Canada, France, Japan, the Republic of Korea, the UK and the USA. In 2003, two more countries joined: South Africa and Switzerland. In 2006, EURATOM, China and the Russian Federation joined.

[8] As of 2007, there are 3 technical working groups still active dealing with economics, risk and safety, and proliferation resistance and physical protection.

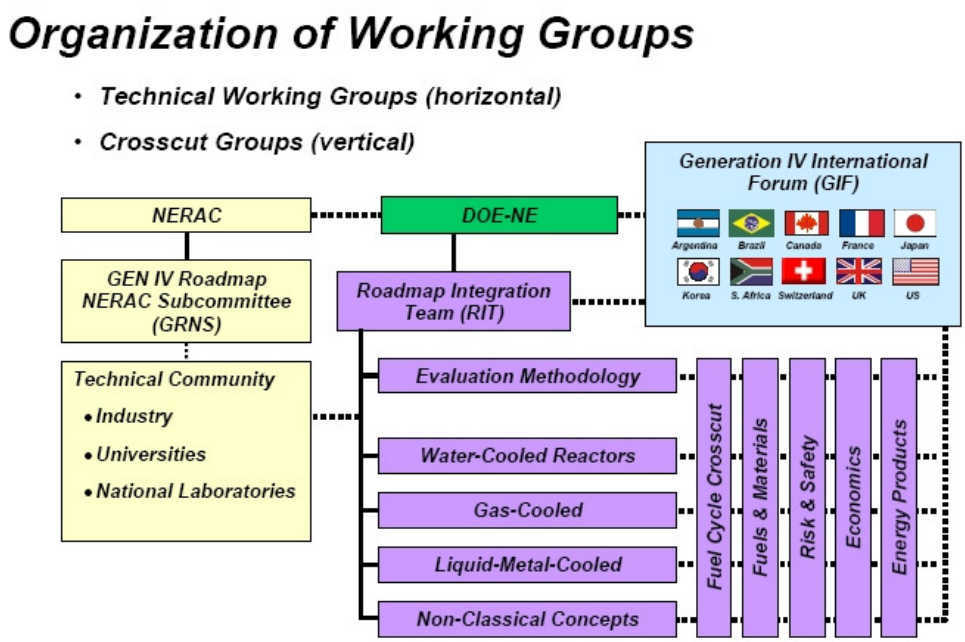


FIG. 2.65. Generation IV working groups and their targets.

GIF operations are overseen by a Policy Group and supported by an Experts Group. The Policy Group acts as a decision making body for high level initiatives and issues, while the Experts Group oversees the various cross-cutting and methodology collaborations. Both groups include representatives from all members.

The GIF has no permanent facilities, budget or staff; members contribute staff time and the use of facilities as needed. The OECD Nuclear Energy Agency serves as a Technical Secretariat to the GIF.

Beginning in 2001, over 100 experts from 10 countries and international organizations began work on defining the goals for new systems, identifying many promising concepts, and evaluating them, and defining the R&D needed for the most promising systems.

Also right from the beginning, GIF members sought to have the most flexible working arrangements to define Generation IV systems and their R&D. As with many other large scale R&D initiatives, the GIF adopted a roadmapping approach to setting goals for Generation IV nuclear energy systems. The results of these efforts were updated several times per year and a 'final' Technology Roadmap for Generation IV for Nuclear Energy Systems has been achieved by the end of 2002 [38]. The time horizon extends to 2030 or earlier.

GIF identified the following reactor systems as of potential importance for the future of nuclear energy systems:

- Gas cooled fast reactor system (GFR) (Fig. 2.66);
- Lead cooled fast reactor system (LFR) (Fig. 2.67);
- Molten salt reactor system (MSR) (Fig. 2.68);
- Sodium cooled fast reactor system (SFR) (Fig. 2.69);
- Supercritical water cooled reactor system (SCWR) (Fig. 2.70);
- Very high temperature reactor system (VHTR) (Fig. 2.71).

The six systems feature increased safety, improved economics for electricity production and new products such as hydrogen for transportation applications, reduced nuclear wastes for disposal, and increased proliferation resistance.

Generation IV nuclear energy systems are future, next generation technologies that will compete in all markets with the most cost-effective technologies expected to be available over the next three decades.

Comparative advantages include reduced capital cost, enhanced nuclear safety, minimal generation of nuclear waste, and further reduction of the risk of weapons materials proliferation. Generation IV systems are intended to be responsive to the needs of a broad range of nations and users.

In addition to selecting these six concepts for deployment between 2010 and 2030, the GIF recognised a number of (international near-term deployment) advanced reactors available before 2015.

Most of the six systems employ a closed fuel cycle to maximise the resource base and minimise high level wastes to be sent to a repository. Three of the six are fast reactors and one can be built as a fast reactor, one is described as epithermal, and only two operate with slow neutrons like today's plants.

Only one is cooled by light water, two are helium cooled and the others have a lead–bismuth, sodium or fluoride salt coolant. The latter three operate at low pressure, with significant safety advantage. The last one has uranium fuel dissolved in the circulating coolant. Temperatures range from 510°C to 1000°C, compared with less than 330°C for today's light water reactors, and this means that four of them can be used for thermo-chemical hydrogen production.

The sizes range from 150 to 1500 MW(e), with the lead cooled one optionally available as a 50–150 MW(e) nuclear battery with a long core life (15–20 years without refuelling) as replaceable cassette or entire reactor module. This concept is designed for distributed generation or desalination.

At least four of the systems have significant operating experience already in most respects of their design, which may mean that they can be in commercial operation well before 2030.

Currently, the thirteen members of the GIF (Argentina, Brazil, Canada, China, Euratom, France, Japan, the Republic of Korea, the Russian Federation, South Africa, Switzerland, the United Kingdom, and the United States of America) are working together as the Generation IV International Forum (GIF) to lay the groundwork for the fourth generation of nuclear energy systems — Generation IV.

The 6 reactor systems have been categorized with regard to various basic goals of energy conversion:

- Energy products, i.e. to produce either electricity or process heat, in particular to produce hydrogen, or both;
- Types of fuel cycles, to be operated in a once-through fuel cycle or with actinide management to a best case deployment date (Table 2.16).

TABLE 2.16. GENERATION IV BEST CASE SCHEDULES FOR DEPLOYMENT

Generation IV	Best case deployment
System	Date
SFR	2015
VHTR	2020
G FR	2025
MSR	2025
SCW R	2025
LFR	2025

For costs of R&D look into the report called A Technology Roadmap for Generation IV Nuclear Energy Systems [38]. It has to be stated here, that these schedules for deployment, even if they are called Best Case Data appear more than optimistic for most of the envisaged advanced systems, except for the SFR, where, as already mentioned above, there exists already quite some experience in several countries involved in or at least cooperating with GEN IV.

However, for each of these six reactors systems reactor parameters and reference values have been defined. Necessary R&D is discussed, including R&D crosscutting between the individual reactor concepts. Schedules and cost are being estimated.

In the following text schematic figures and some technical information are given for the six selected reactor types.

Gas cooled fast reactor (GFR)

This type (Fig. 2.66, Table 2.17) features a fast neutron spectrum, helium cooled reactor with a closed fuel cycle.

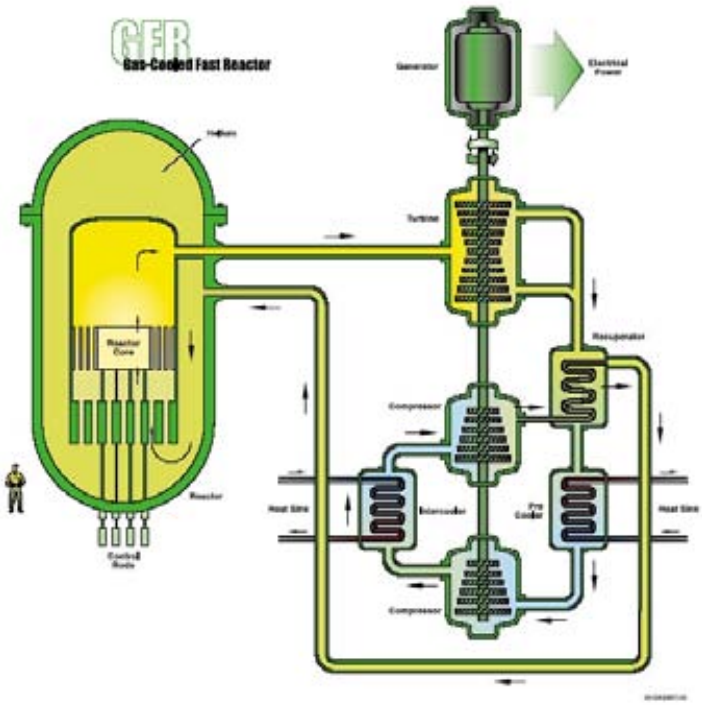


FIG. 2.66. Generation IV gas cooled fast reactor system (GFR).

Like other helium cooled reactors which have operated or are under development, these will be high temperature units (~850°C), suitable for power generation, thermo-chemical hydrogen production or other process heat. For electricity, the gas will directly drive a gas turbine (Brayton cycle). Fuels would include depleted uranium and any other fissile or fertile materials. Spent fuel would be reprocessed on-site and all the actinides recycled to minimise production of long lived radioactive wastes. While General Atomics worked on the design in the 1970s (but not as fast reactor), none has so far been built.

TABLE 2.17. GAS COOLED FASTS REACTOR: REACTOR PARAMETERS AND REFERENCE VALUES

Reactor parameters	Reference value
Reactor power	600 to 3000 MW(th) (300 to 1500 MW(e))
Net plant efficiency (direct cycle helium)	>45%
Coolant inlet/outlet temperature	Up to 850°C
Core pressure drop/primary coolant pressure	Consistent with gas natural circulation / 5–7 MPa
Average power density	50–100 MW(th)/m^3
Reference fuel compound	(U, Pu, MA) – C, or N or O_2
Candidate fuel concepts	Dispersion fuel, solid solution, particle fuel
Volume fraction, fuel/gas/SiC	50/40/10%
Conversion ratio	Self-sustaining core
Burnup, damage	~10 at%

Lead cooled fast reactor (LFR)

This type (Fig. 2.67, Table 2.18) features a fast spectrum, lead or lead–bismuth eutectic liquid metal cooled reactor, and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides.

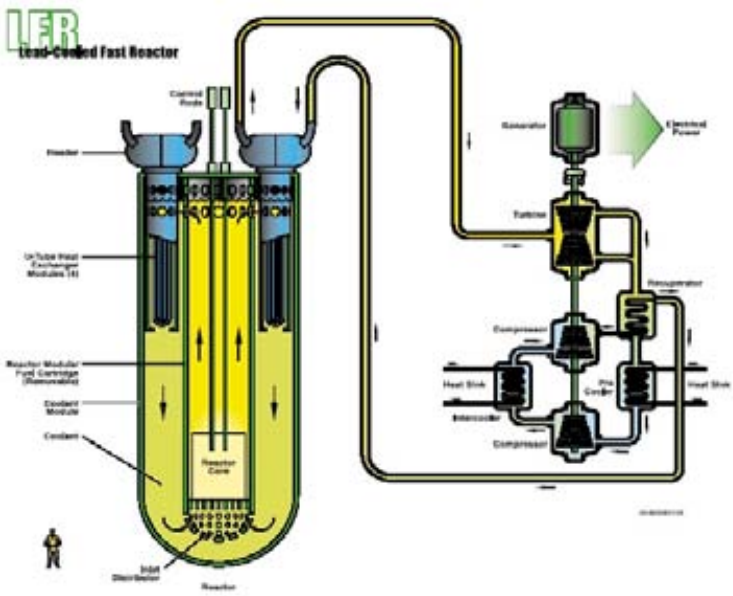


FIG. 2.67. Generation IV lead cooled fast reactor system (LFR).

Liquid metal (Pb or Pb–Bi) cooling is by natural convection. Fuel is depleted uranium metal or nitride, with full actinide recycling from regional or central reprocessing plants. A wide range of unit sizes is envisaged, from factory-built nuclear battery with 15–20 year life for small grids or developing countries, to modular 300–400 MW(e) units and large single plants of 1400 MW(e). Operating temperature of 550°C is readily achievable but 800°C is envisaged with advanced materials and this would enable thermo-chemical hydrogen production.

TABLE 2.18. LEAD COOLED FAST REACTORS: REACTOR PARAMETERS AND REFERENCE VALUES

Reactor parameters	Reference values			
	Pb–Bi battery (nearer-term)	Pb–Bi module (nearer-term)	Pb large (nearer-term)	Pb battery (far-term)
Coolant	Pb–Bi	Pb–Bi	Pb	Pb
Outlet temperature (°C)	~550	~550	~550	~750–800
Pressure (at)	1	1	1	1
Rating (MW(th))	125–400	~1000	3600	400
Fuel	Metal or Nitride	Metal Alloy	Nitride	Nitride
Cladding	Ferritic	Ferritic	Ferritic	Ceramic coatings or refractory alloys
Average burnup (GWD/MT_{HM})	~100	~100–150	100–150	100
Conversion ratio	1.0	d ≥1.0	1.0–1.02	1.0
Lattice	Open	Open	Mixed	Open
Primary flow	Natural	Forced	Forced	Natural
Pin liner heat rate	Derated	Nominal	Nominal	Derated

This design concept corresponds to the Russian Federation's BREST fast reactor technology which is lead cooled and builds on 40 years experience of lead–bismuth cooling in submarine reactors. Its fuel is U + Pu nitride. The GIF proposal appears to arise from two experimental designs: the US STAR and Japan's LSPR, these being lead and lead–bismuth cooled respectively.

Molten salt reactor (MSR)

This type (Fig. 2.68, Table 2.19) produces fission power in a circulating molten salt fuel mixture with an epithermal-spectrum reactor and a full actinide recycle fuel cycle.

The uranium fuel is dissolved in the sodium fluoride salt coolant which circulates through graphite core channels to achieve some moderation and an epithermal neutron spectrum. Fission products are removed continuously and the actinides are fully recycled, while plutonium and other actinides can be added along with ^{238}U. Coolant temperature is 700°C at very low pressure, with 800°C envisaged. A secondary coolant system is used for electricity generation, and thermo-chemical hydrogen production is also feasible.

During the 1960s, the USA developed the molten salt breeder reactor as the primary backup option for the conventional fast breeder reactor and a small prototype was operated. Recent work has focused on lithium and beryllium fluoride coolant with dissolved thorium and ^{233}U-fuel. The attractive features of the MSR fuel cycle include: the high level waste comprising fission products only, hence shorter lived radioactivity; small inventory of weapons-fissile material (^{242}Pu being the dominant Pu isotope); low fuel use (the French self-breeding variant claims 50 kg of thorium and 50 kg ^{238}U per billion kW·h); and safety due to passive cooling up to any size.

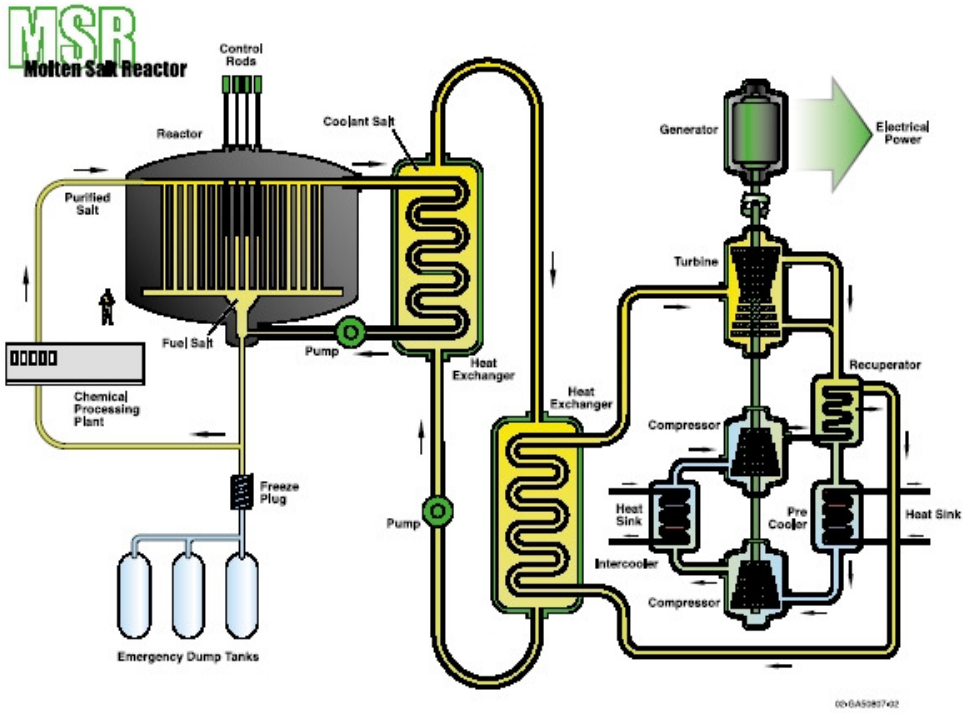


FIG. 2.68. Generation IV molten salt reactor system (MSR).

TABLE 2.19. MOLTEN SALT REACTORS, REACTOR PARAMETERS AND REFERENCE VALUES

Reactor parameters	Reference value
Net power	1000 MW(e)
Power density	22 MW(th)/m^3
Net thermal efficiency	44 to 50%
Fuel-salt – inlet temperature – outlet temperature – vapour pressure	565°C 700°C (850°C for hydrogen production) < 0.1 psi
Moderator	Graphite
Power cycle	Multi-reheat recuperative helium Brayton cycle
Neutron spectrum burner	Thermal-actinide

Sodium cooled fast reactor (SFR)

This type (Fig. 2.69, Table 2.20) features a fast spectrum, sodium cooled reactor with a closed fuel cycle for efficient management of actinides and conversion of fertile uranium.

This builds on more than 300 reactor-years experienced with fast neutron reactors over five decades and in eight countries. It utilises depleted uranium in the fuel and has a coolant temperature of 550°C enabling electricity generation via a secondary sodium circuit, the primary one being at near atmospheric pressure. Two variants are proposed: a 150 to 500 MW(e) type with actinides incorporated into a metal fuel requiring pyrometallurgical processing on-site, and a 500–1500 MW(e) type with conventional MOX fuel reprocessed in conventional facilities elsewhere.

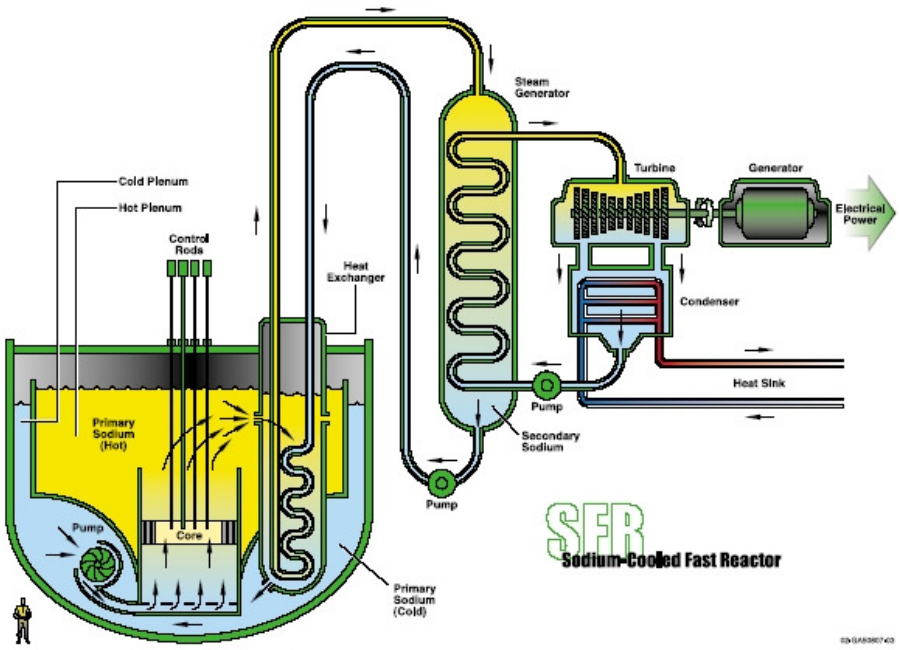


FIG. 2.69. Generation IV sodium cooled fast reactor system (SFR).

TABLE 2.20. SODIUM COOLED FAST REACTOR: REACTOR PARAMETERS AND REFERENCE VALUES

Reactor parameters	Reference value
Outlet temperature	530–550°C
Pressure	~1 Atmospheres
Rating	1000–5000 MW(th)
Fuel	Oxide or metal alloy
Cladding	Ferritic or ODS ferritic
Average burnup	~150–200 GWD/MT_{HM}
Conversion ratio	0.5–1.30
Average power density	350 MW(th)/m^3

Supercritical water cooled reactor (SCWR)

This type (Fig. 2.70, Table 2.21) is a high temperature, high pressure water cooled reactor that operates above the thermodynamic critical point of water to give a thermal efficiency about one third higher than today's light water reactors from which the design evolves.

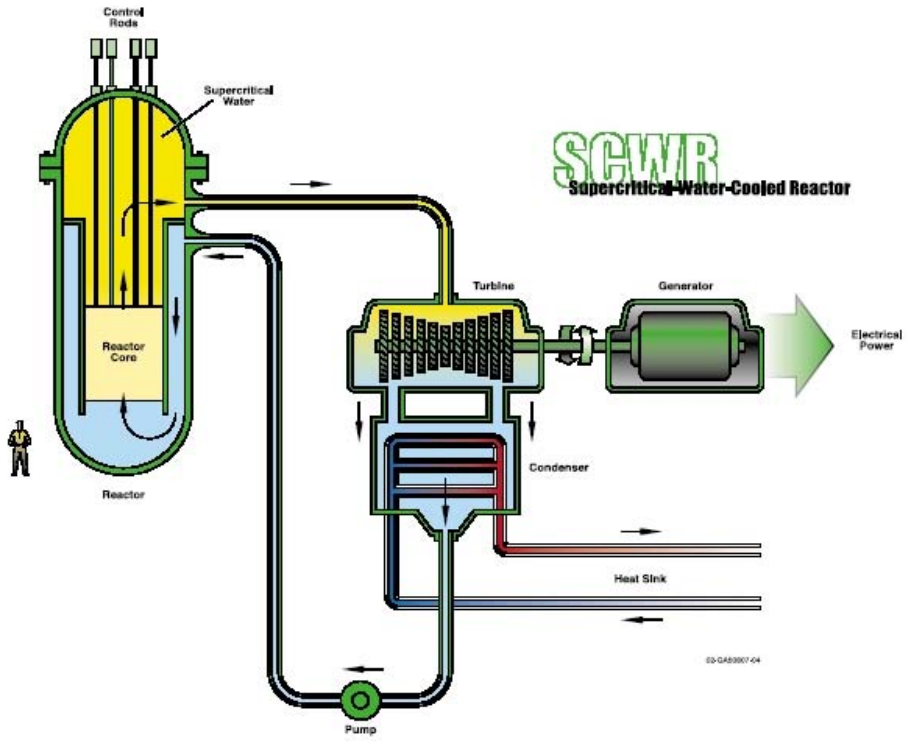


FIG. 2.70. Generation IV supercritical water cooled reactor system (SCWR).

The supercritical water (25 MPa and 510–550°C) directly drives the turbine, without any secondary steam system. Passive safety features are similar to those of simplified boiling water reactors. Fuel is uranium oxide, enriched in the case of the open fuel cycle option. However, it could be built also as a fast reactor with full actinide recycling based on conventional reprocessing. Most research on this design has been done in Japan.

TABLE 2.21. SUPERCRITICAL WATER COOLED REACTOR: REACTOR PARAMETERS AND REFERENCE VALUES

Reactor parameters	Reference value
Plant capital cost	US $900/KW
Unit power and neutron spectrum	1700 MW(e), thermal spectrum
Net efficiency	44%
Coolant inlet and outlet temperatures and pressure	280°C/510°C 25 MPa
Average power density	~100 MW(th)/m^3

Very high temperature reactor (VHTR)

This type (Fig. 2.71, Table 2.22) is a graphite-moderated, helium cooled reactor with a once-through uranium fuel cycle;

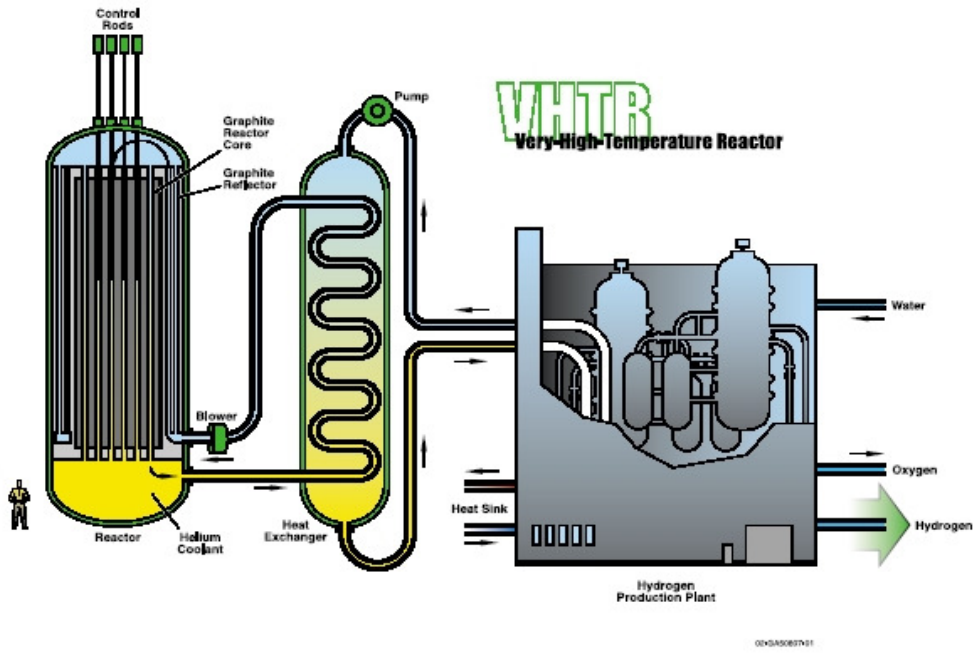


FIG. 2.71. Generation IV very high temperature reactor system (VHTR).

These are graphite-moderated, helium cooled reactors, based on substantial experience. The core can be built of prismatic blocks such as the Japanese HTTR and the GT-MHR under development by General Atomics and others in the Russian Federation, or it may be a pebble bed such as the Chinese HTR-10, and the PBMR under development in South Africa with international partners. Outlet temperature of 1000°C enables thermo-chemical hydrogen production via an intermediate heat exchanger, with electricity cogeneration, or direct high efficiency driving of a gas turbine (Brayton cycle). There is some flexibility in fuels, but no recycling. Modules of 600 MW thermal are envisaged.

TABLE 2.22. VERY HIGH TEMPERATURE REACTOR: REACTOR PARAMETERS AND REFERENCE VALUES

Reactor parameters	Reference value
Reactor power	600 MW(th)
Coolant inlet/outlet temperature	640/1000°C
Core inlet/outlet pressure	Dependent on process
Helium mass flow rate	320 kg/s
Average power density	6–10 MW(th)/m^3
Reference fuel compound	ZrC-coated particles in blocks, pins or pebbles
Net plant efficiency	>50%

The international effort of GIF reached a major milestone on 28 February 2005, as five of the forum's member countries signed the world's first agreement aimed at the international development of advanced nuclear energy systems, the Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy Systems.

As of January 2007, this agreement was in force for the following countries:

TABLE 2.23. COUNTRIES THAT SIGNED THE GIF R&D FRAMEWORK AGREEMENT

Country	Implementing agent(s)
Canada	Department of Natural Resources
Euratom	Joint Research Centre
France	Commissariat à l'Énergie Atomique
Japan	Agency for Natural Resources and Energy Japan Atomic Energy Agency
Republic of Korea	Ministry of Science and Technology (MOST) Korea Science and Engineering Foundation (KOSEF)
Switzerland	Paul Scherrer Institute
United States of America	Department of Energy

An important milestone for the GIF was reached in March 2007 with the signature of the first Project Arrangement, regarding Advanced Fuel for the SFR System. Signed by the five partners to the SFR System Arrangement, the project agreement sets out a detailed plan for research and development activities in this area and details the schedule, funding and deliverables expected to achieve this. Other project arrangements are expected to be signed soon.

2.6.2. Global nuclear energy partnership (GNEP)

In February 2006, the US government announced an initiative called GNEP [40]. The initiative proposes 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 [41], 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) reactors (ABR) [42] to reduce actinides in waste, and the reduction of separated civilian plutonium.

However, the GNEP concept also includes some political aspects: It proposes to confine the international fuel service market (see Ref. [43]). Primarily, the Nuclear Weapons States i.e. the USA, the UK, France, the Russian Federation and China and some additional nations to be selected (like Japan) would become Fuel Cycle Nations supplying nuclear fuel to the rest of the nuclear community called Reactor Nations. Receiving fuel would require the Reactor Nations committing to forgo national enrichment and reprocessing activities. Spent fuel would have to be sent back to the Fuel Cycle Nations, only who would be entitled to recycle and reuse all materials such as Pu.

As of September 2007, sixteen countries have signed a Statement of Principles: China, France, Japan, the Russian Federation, and the USA as the original partners, of GNEP, and Australia, Bulgaria, Ghana, Hungary, Jordan, Kazakhstan, Lithuania, Poland, Romania, Slovenia, and Ukraine. These principles promote cooperation on a series of key issues:

- the need to deal with waste materials in a responsible manner;
- the costs involved with developing the necessary infrastructure;

- the need to develop and deploy technologies that will increase the efficiency of the fuel cycle;
- the risks posed by the potential for proliferation of nuclear materials and sensitive technologies.

GNEP will move the USA from a once-through to a closed or recycling fuel cycle. Three new nuclear facilities are to be developed in the USA: A nuclear fuel recycling centre (CFTC) reprocessing spent fuel from LWRs, an advanced fuel cycle facility (AFCF) producing fuel with transuranics added, and the advanced recycling reactor (ABR) burning the transuranics while generating electricity.

2.6.3. The EURATOM research on future nuclear systems

The European Commission funds studies and investigations on the future of nuclear systems in accordance with a Green Paper on the security of energy supply in Europe presented by the European Commission in 2000 [44]. Those studies focus on future systems for:

- energy production;
- waste management such as partitioning and transmutation (P&T).These activities mainly involve the EURATOM research centres and their activities are as follows:

- Institute for Transuranium Elements (ITU) – Karlsruhe:
  - P&T (partitioning techniques and actinide fuel for transmutation);
  - improved and safer fuel fabrication routes.
- Institute for Energy (IE) – Petten:
  - actinide and innovative fuel irradiation in HFR Networks.
- Institute for Reference Materials and Measurements (IRMM) – Geel:
  - neutron data for transmutation.

Work is performed in cooperation with the nuclear industry, other research centres (e.g. ORNL/USA studies on molten salt reactor (MOST)), and Universities (e.g. Tokyo University studies on high performance light water reactors (HPLWR), operating in thermodynamically supercritical regime).

The European Commission (EC) has joint the Generation IV International Forum in 2006. Within the 6th Framework Programme (FP6)[9], the EC initiated a research programme on gas cooled fast reactors called GCFR STREP, lead cooled reactors (ELSY), and very high temperature reactors (e.g. RAPHAEL, PUMA).

2.6.4. The USA–Russian Federation collaborative programme

In the Russian Federation, many studies and research activities are underway to investigate the potential and to prepare the future of advanced reactor systems. Some of them are now (since 2002) being performed in collaboration with the USA [27].

This collaboration includes:

- fast liquid metal cooled reactors (with Na, Pb + Bi, Pb only);
- fast and thermal high temperature He cooled reactors;
- reactors fuelled with molten salts;
- light water reactors with supercritical coolant parameters.

[9] See web site: http://cordis.europa.eu/fp6-euratom.

There is also collaboration on various fuel cycle alternatives and special waste management methods, which will be discussed later.

The Russian Federation is also a strong supporter of the programme The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) which is in progress since 2002 under the umbrella of the IAEA.

2.7. Very advanced reactor systems

There are very ambitious systems under consideration, some of them already under first investigations that are today far away from becoming commercial. In fact, even the development of prototypes is still rather far away. However, there are concepts under consideration with very promising perspectives for the future of nuclear energy and its political and social acceptance.

Those concepts promise substantial progress with regard to the present long term burden on the final disposal of waste and/or spent fuel. And they also promise serious progress with the non-proliferation issue. Interestingly, those concepts are, historically speaking, not fully new. In the very early days of nuclear energy there were already all basic ideas available, however the existing possibilities of the technology and also the political and market requirements just did not provide realistic chances for those ideas. This holds true in particular for accelerator driven systems (ADS), and for other highly efficient (actinide) burners and/or breeders, like the molten salt reactor (MSR) concept. In some concepts both systems are combined.

2.7.1. Accelerator driven systems (ADS)

The idea of this concept is to shoot highly accelerated particles (e.g. protons) at a heavy material target. This bombardment produces a very intense neutron source (a process called spallation). These neutrons enter a subcritical core (also called a blanket) where they can be multiplied.

With these neutrons Pu, for example from nuclear weapons, could be burnt, a U–Th fuel cycle may be started, and very long living radioactive isotopes like the minor actinides can be transmuted by fission into short living less heavy isotopes. Thus, there are several very promising perspectives for the development of such a system. Many conceptual studies and also experimental investigations are presently underway worldwide (to be discussed further in Section 5.4). However, there is still a long — and costly — way to go to end up with a commercially applicable solution.

2.7.2. The molten salt reactor (MSR)

As all recently returning advanced concepts, also this one has a long history. The first reactor of this kind was the 2.5 MW(th) aircraft reactor experiment that in 1954 demonstrated maximum operating temperatures of up to 860ºC. This was followed, in the 1960s, by the molten salt reactor experiment, an 8 MW(th) reactor to investigate for the first time key features for a molten salt breeder reactor (MSBR). This is the rather old but anyway existing experience basis for the today again rising interest in this reactor concept due to its attractive perspectives for advanced fuel cycles.

Molten salt reactors (MSR) can be used for production of electricity, actinide burning, production of hydrogen, and production of fissile fuels. Concept development studies have been performed in various places, like in France [45], the USA [46, 47], and the Russian Federation.

Fissile, fertile and fission products are dissolved in a high temperature molten fluoride salt with a very high boiling point (1400ºC) that is both the reactor fuel and the coolant. The reactor, in principle, can be built in large sizes with passive safety systems (Fig. 2.72).

This reactor concept features unique possible applications [47]:

- destruction of long lived radionuclides of a wide range of possible fuel cycles (once-through, waste burning, fissile fuel production, e.g. in a thorium uranium fuel cycle) without major changes in the reactor design;
- limiting the radioactivity in the reactor core by on line removal and solidification of the mobile fission products;
- limited excess reactivity requirements in the core due to online fuel management.

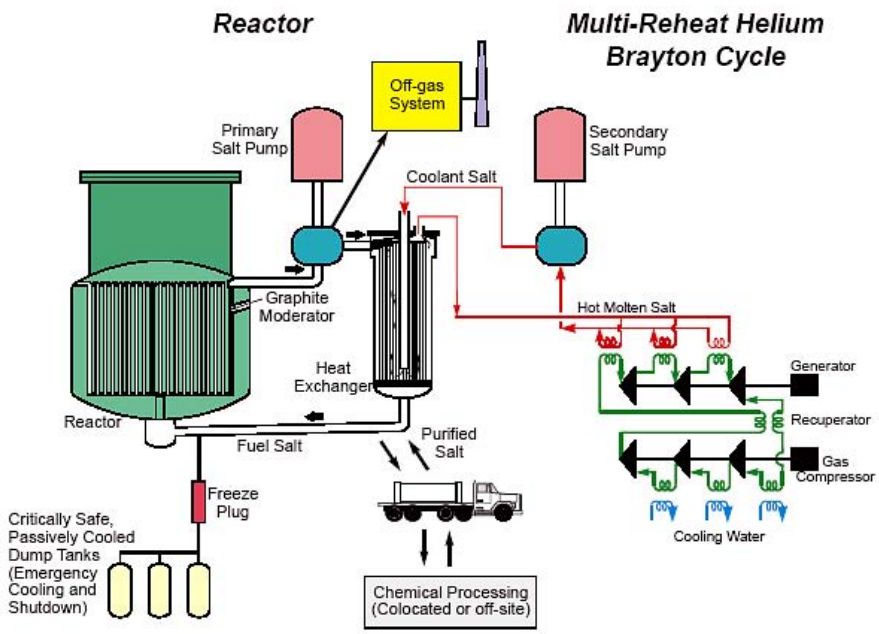


FIG. 2.72. Molten salt reactor [46].

2.7.2.1. French studies of MSR

Because of the particularly attractive perspectives of deploying a MSR system within a Th/U fuel cycle, providing the possibilities of using the large natural recourses of thorium, as a specific example the concept of the reactor physics group of the Laboratoire de Physique Subatomique et de Cosmologie at Grenoble in France is described shortly in the following paragraphs.

The core of the reactor is cylindrical (Fig. 2.73). It is a block of graphite with holes in which the molten salt circulates. The graphite serves as a neutron moderator and is the solid structure of the reactor. The molten salt circulating in the channels is both the fuel and the coolant. It contains the thorium and uranium needed to sustain the chain reaction.

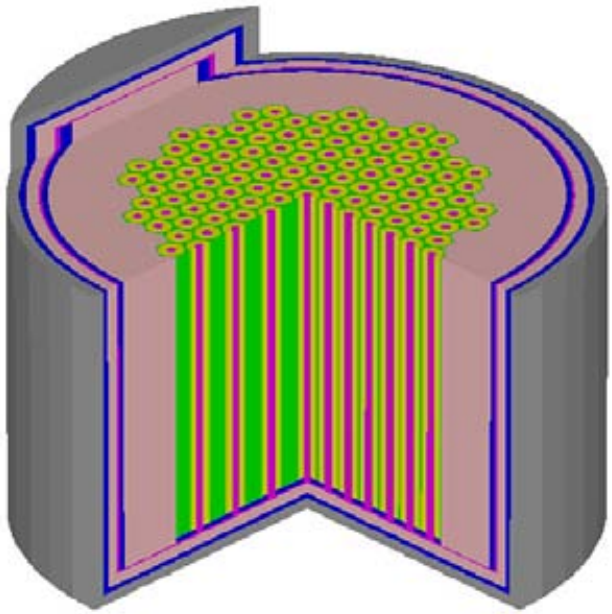


FIG. 2.73. View of a molten salt reactor model.

In a simulated reactor model providing 2500 MW(th) thermal power, equivalent to 1000 MW(e) electric power, at the chosen operating temperature (salt temperature) of approximately 700°C in the heat exchanger, the thermal efficiency gets up to 40%. The carrier salt is a mixture of lithium fluoride (LiF) and of beryllium fluoride (BeF_2), or 4LiF-BeF_2, to which the fuel is added. The salt passes into a tank above the core to facilitate salt circulation, and from there to the heat exchanger. The salt, cooled to approximately 600°C, returns to the tank below the core, from which it enters into the core.

The on-line chemical processing takes the salt through the following steps:

- Extract the uranium and put it back in the reactor.
- Extract the protactinium, let it sit for three months and, once it has decayed to uranium, put it back in the circuit.
- Extract the fission products.
- Replace the thorium consumed.

If the reactor operates as a breeder, extract the excess uranium.

An important feature is the fact that all heavy nuclei (uranium and transuranics, called minor actinides) remain in the reactor (except some leaking out during processing). Because they remain in the salt indefinitely, almost all heavy nuclei will eventually fission. The actinides produced in the reactor are thus incinerated in the reactor. The only radioactive nuclei that escape the reactor are the fission products. In Fig. 2.74 the schematics of the fuel processing are shown.

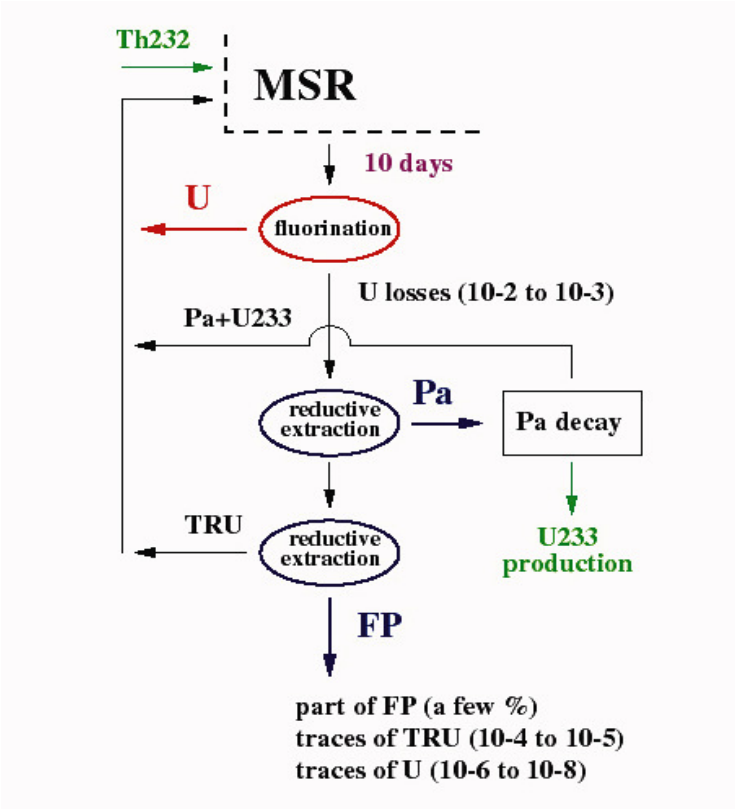


FIG. 2.74. Schematic of the fuel processing system. (TRU: transuranics; MSR: molten salt reactor; FP: fission products)

A thorium-based molten salt reactor cannot be started without fissile material. The performed simulations have shown that plutonium taken from a PWR's spent UO_2 fuel can be used to start a thorium-based molten salt reactor. As the plutonium is incinerated, it is progressively replaced with ^{233}U. The irradiation of plutonium generates more minor actinides than that of ^{233}U. These are progressively incinerated in the reactor but it takes more than a hundred years to reduce the minor actinides to the amount in a Th–^{233}U reactor that was started directly with ^{233}U. Such a reactor, with its on line fuel processing, is at least a converter. Thus, it burns ~100% of its fuel.

Reactor safety has not been examined in detail at this stage (2001). All the simulations apply to a critical reactor but at least during the testing phase of an experimental reactor, and perhaps finally, it could be operated as a subcritical reactor, associated to a proton accelerator to produce the missing neutrons via a spallation reaction. A subcritical reactor can stay safe even if the effective multiplying coefficient, k_{eff}, changes, within limits.

This study shows, as an example of many more others presently performed, the large promising potential of the concept. But it also shows how far this interesting concept still is from realization. And it has to be clear that this reactor concept faces extreme material problems, due to the very aggressive liquid medium at very high temperatures. The graphite may be able to resist to this highly aggressive medium. In the chemical industry there is experience with similar media but not at these high temperatures. And if materials are found withstanding for a reasonable time, the cost of those materials and their replacement from time to time may raise immense economic obstacles.

2.7.2.2. Russian studies of the MSR

The following Russian contribution describes some of the problems involved with the molten salt reactor concept in more detail.

A major liquid salt reactor (LSR) development problem is the selection of the fuel salt. The major criteria for this selection are:

- low neutron absorption cross-sections of the salt components;
- sufficient solubility of fuel components at acceptable operation temperatures;
- stability in the accumulation of difficult to remove fission products;
- radiation and chemical stability of the salt (no decomposition, no low volatility of components);
- adequate hydrodynamic and thermal characteristics;
- possibility for effective fuel recycling for FP and PK (economic reprocessing).

The range of nuclides with a small neutron capture cross-section that could be potentially considered as coolant (fuel) components is rather broad, but different combinations of four fluorides (^{7}LiF, BeF_2, ZrF_4, NaF) have been largely considered in liquid-salt systems for already 50 years.

The basic fuel compositions studied in the LSR development are the following (% mol.):

- ARE – 53,0 NaF; 41,2 ZrF_4; 5,8% UF_4;
- MSRE – 65,0 LiF; 29,1 BeF_2 ; 5,0 ZrF_4; 0,9 UF_4;
- MSBR – 71,772 LiF; 16,0 BeF_2; 12,0 ThF_4; 0,228 UF_4.

As the NaF-ZrF_4 system was used in the first LSR, the conversion to the LiF-BeF_2 composition was mainly connected with the development of breeder reactors for which the neutron economy is a top priority issue.

The major drawbacks of the LiF-BeF_2 systems are:

- Generation of tritium in neutron reactions of lithium even with removal of a large portion of the ^{6}Li isotope.
- High cost of the ^{7}Li isotope and beryllium.
- High toxicity of beryllium. The permissible content of Be in a production room air is equal to 1.10^{-4} mg/m^{3}.

Drawbacks of NaF-ZrF_4 are:

- high pressure of zirconium tetra-fluoride vapours;
- a higher melting temperature (732 K) than in the lithium–beryllium composition.

The selection requirements of fuel salt for dissolution of irradiated fuel elements and their subsequent gas-fluoride reprocessing with production of highly purified uranium are:

- low cost of the fluoride system components for dissolution of irradiated fuel elements;
- low melting temperature (less than 550–600 C);
- low viscosity;
- small vapour pressure at working temperatures.

The mixtures of the LiF-BeF_2, KF-LiF, KF-ZrF_4, NaF-ZrF_4, KF-ZrF_4-LiF, KF-ZrF_4-AlF_3 salts meet these conditions.

Uranium–zirconium fuel elements are dissolved in these melts by bubbling through fluorine hydrogen. Uranium and zirconium pass into a highly soluble form of UF_4 and ZrF_4, and noble gases and highly volatile FP pass into a gas phase and are delivered to the gas treatment

system. To recover the LSR waste fluorine melts, there were used the $NaF-ZrF_4$ and $LiF-BeF_2$ systems that were fluorinated at 550 C with the subsequent UF_6 purification by a sorption method. The existing structural materials limit the maximum working FS temperature with a maximum of ~750 C.

But it is not the eutectics temperature value that is of importance for the practical selection of the fluoride system, i.e. the lowest melting temperature, but the actual temperature, with which the dissolution of a large quantity of actinide fluorides can be achieved, and the width of the fluoride composition concentration range, which includes the system in a molten state. None of the most advanced salt compositions (LiF-NaF) with a melting temperature of 650 C meets this criterion.

Since some FP fluorides (CeF_3, BaF_2, SrF_2) in the melt reduce the solubility of actinides, actually for maintaining all actinides, lanthanides and other fission products present in the irradiated product in a solved state, the fluoride solution should have the working temperature of 600–700 C. The americium and curium solubility can be roughly assessed based on the solubility data of rare-earth metals (REM), i.e. their close chemical analogs.

The $NaF-ZrF_4$ system has advantages over the $LiF-BeF_2$ system in terms of the solubility of both actinides and lactinides. This factor is important as both the LSR life and the irradiated fuel reprocessing cost depend on it. To achieve a high UF_4, PuF_3 and REM fluoride solubility, the $3LiF-BeF_2$ system should be used whose melting temperature (540 C) is higher than the melting temperature of the $NaF-ZrF_4$ system (510 C). However, the total cost of the LiF- BeF_2 composition is approximately 100 times as high as the $NaF-ZrF_4$ cost.

An additional drawback of the $LiF-BeF_2$ system is a very high hygroscopicity of BeF_2, which causes difficulties in the production, use and recovery of irradiated fuel.

For a fuel based on the $NaF-ZrF_4$ melt, the recovery technology can be simplified with respect to the removal of REM fluorides having high neutron entrapment cross-sections and reducing the MA solubility. The technology of purifying the spent working melt is reduced to its cooling to a temperature at which only REM fluorides would be separated. Thereafter, high temperature filtration or centrifugation and the sludge washing with a fresh $NaF-ZrF_4$ melt will be required. The residue after the washing is supplied for the controlled storage and the melt purified of REMs is returned to the reactor or for dissolution of the irradiated fuel. The MA behaviour during this operation requires additional studies.

The final selection of the transmutation LSR FS primarily requires studies of the plutonium and MA solubility in the candidate FS. The problem can be solved also through development of new structural materials that raise the temperature ceiling and, thus, the solubility of transmuted elements in the FS. Of major importance for LSR as a reactor with a closed fuel cycle is the maximum economic FS reprocessing. Less operations, plutonium and MA inventories in the reprocessing cycle and secondary waste amounts do not only reduce the cost of the TS reprocessing and RW disposal, but also the risk caused by occurrence of any emergency situations.

2.7.2.3. US studies of MSR

Currently, within the Generation IV project [48] a concept of a modular MSR with TRISO fuel is studied in the USA, called AHTR.

2.8. Perspectives of nuclear power

There is no easy answer to the question how the use of nuclear energy will develop during the next decades, regionally and worldwide. Certainly the energy demand will grow worldwide

due to the urgent need in many areas of this world to improve living conditions and also due to the fact that the world population is still growing. Many prognostic scenarios were elaborated on the quantitative development of this growth during the last 50 years. Some of them were wrong, in particular with regard to the regional situation. Therefore, 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 (Fig. 2.75) as for example the evaluation of SRES scenarios by the IAEA, performed within INPRO [1].

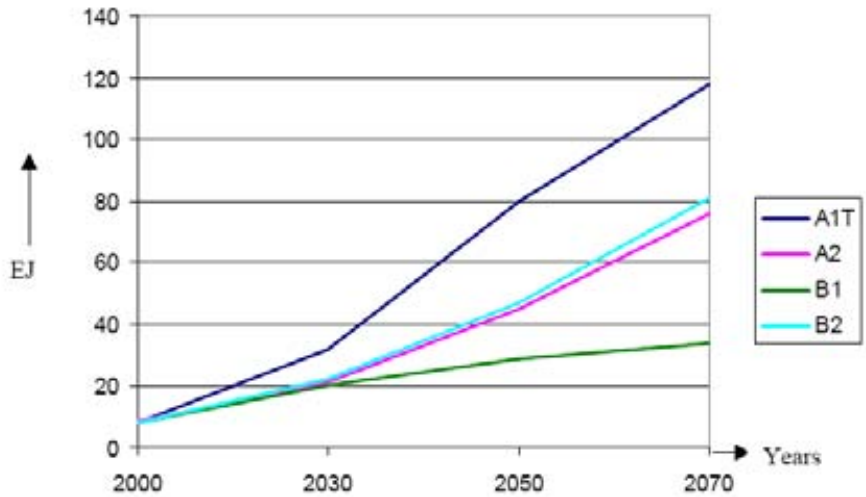


FIG. 2.75. Nuclear electricity production (EJ) for four selected SRES scenarios A1T, A2, B1 and B2 (Section 4.1.1 of Ref. [1]).

The four selected scenarios in Fig. 2.75 are characterized as follows:

- The A1T storyline and scenario family describe a future world of very rapid economic growth, low population growth, and the rapid introduction of new and more efficient technologies. Major underlying themes are convergence among regions, capacity building, and increased cultural and social interactions, with a substantial reduction in regional differences in per capita income.
- The A2 storyline and scenario family describe a very heterogeneous world. The underlying theme is self-reliance and preservation of local identities. Fertility patterns across regions converge very slowly, which results in high population growth. Economic development is primarily regionally oriented and per capita economic growth and technological changes are more fragmented and slower than in other storylines.
- The B1 storyline and scenario family describe a convergent world with the same low population growth as in the A1 storyline, but with rapid changes in economic structures toward a service and information economy, with reductions in material intensity, and the introduction of clean and resource-efficient technologies. The emphasis is on global solutions to economic, social, and environmental challenges, including improved equity, but without additional climate initiatives.
- The B2 storyline and scenario family describe a world in which the emphasis is on local solutions to economic, social, and environmental challenges. It is a world with moderate population growth, intermediate levels of economic development, and less rapid and more diverse technological change than in the B1 and A1 storylines. While the storyline is also oriented toward environmental protection and social equity, it focuses on local and regional levels.

All four storylines project a significant total increase of nuclear power worldwide (see Fig. 2.75 above) within the 21st century with significant differences in different areas of the world, i.e. nuclear energy together with other energy sources will be needed to satisfy the future global demand for energy. Some of the SRES scenarios above, however, show stagnation or decline of the worldwide role of nuclear power after about 2080 based on economic assumptions, e.g. solar energy would become cheaper than nuclear energy and therefore the main energy source. However, as also stated in Ref. [1] the share of nuclear power could not only be kept constant but even increased dramatically if its economic competitiveness would be further improved continuously (concept of learning rates) in comparison with alternative energy sources.

Figure 2.76 [5] shows the development of the installed electrical (nuclear) power since its beginning till today and a projection thereof till 2030. By the end of 2007, the installed nuclear capacity amounted to 370 GW(e).

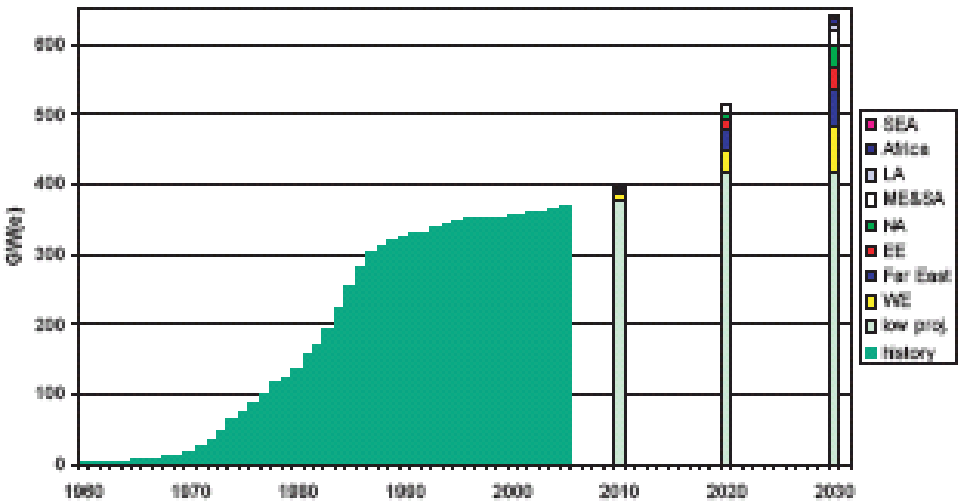


FIG. 2.76. Development of installed nuclear power capacity (GW(e)) worldwide [5].

In Fig. 2.76 above the dark green bars show historical growth from 1960 through 2005. The light green bars show the IAEA's latest low projection for 2010, 2020 and 2030 (420 GW(e)). The other colours show how much of the difference between IAEA's low and high projections (more than 600 GW(e) in 2030) is attributable to different regions of the world (SEA: South-East Asia, LA: Latin America, ME&SA: Middle East and South Asia, NA: North America, EE: Eastern Europe, WE: Western Europe).

For the next two to three decades the light water reactors with further enhanced safety and increased competitiveness will most probably be the worldwide backbone for nuclear power generation, i.e. electricity production. Additionally, it is expected that improved heavy water reactors will contribute to this nuclear share of the overall generation of electric energy.

CHAPTER 3.
NUCLEAR FUEL CYCLE OPTIONS

In this chapter a short overview is laid out on the history of NFC strategies, the current status of NFC technology globally and in some specific countries, and finally the mid term and long term trends of NFC technologies.

3.1. Short history of NFC strategies

Already in the late 1940s and then in the 1950s most of the technological steps of the uranium–plutonium nuclear fuel cycle were developed: mining, milling, conversion, enrichment, fuel fabrication, and recycling of irradiated fuel for recovery of fissile. However, in those times the driving force was the development of nuclear weapons and then the necessity to fuel naval propulsion reactors.

In the 1970s and 1980s, commercial nuclear power was growing rapidly in the United States of America, Japan and Europe (e.g. France, Germany and the United Kingdom).

The fuel cycle strategies in the above mentioned countries changed from the 1970s to the 1980s. In the 1970s, due to oil embargoes the security of energy supply became first priority in most of these countries. Fuel recycling and fast breeder deployment was considered to be of decisive importance. With the declining growth rate of nuclear power, decreasing uranium fuel costs, and technological and operational progress, i.e. increased fuel burnup and load factor in LWRs in the 1980s, however, the thermal reactor with a once-through and MOX mono-recycling NFC became more and more dominant.

On the other hand, there were also political reasons influencing this development more and more strongly as discussed in the following.

In the United States of America, a uranium oxide once-through cycle is currently employed. This is because during the mid-1970s the political leaders in the United States of America concluded that the most severe risks for nuclear proliferation result from fuel reprocessing and recycling. The US administration (president Carter) decided in 1977 to stop commercial reprocessing and recycling of separated plutonium. But also at that time the USA initiated research programmes aiming at more proliferation resistant reprocessing methods capable of recycling all actinides and of maintaining high radiation level at all stages of materials providing a barrier against proliferation.

At the same time (during the 1970s) in several states in Europe (e.g. Belgium, France, Germany and Switzerland) and in Japan, the recycling of Pu in thermal reactor as MOX (mixed U–Pu oxide), was started and till up today developed into a mature commercial technology (Table 3.1), i.e. MOX reaching about the same burnup as pure U fuel. However, it should be mentioned that commercial reprocessing of this kind of fuel has been demonstrated only in France and the UK. Other NFC than the U–Pu fuel cycle have not been deployed commercially so far.

TABLE 3.1. STATUS OF LARGE SCALE MOX FUEL UTILIZATION IN THERMAL REACTORS (2007)

Number of thermal reactors[1]				
	Operating	Licensed to use MOX FAs	Loaded with MOX FAs	Applied for MOX licence
Belgium	7	2	2	
France	58	20	20	5[2]
Germany	17	12	10	4
Japan	55	4	1	1
Switzerland	5	3	3	1
Total	142	41	36	13

Legend:
[1] There are a number of reactors, notably in Europe and India, not included in this table, which are licensed to use MOX fuel on an experimental basis.
[2] Technically capable reactors planned to be licensed.

Japan intends to use MOX in a total of 16 to 18 plants in 2010.

3.2. Current status of NFC technology

Presently, only the raw material uranium is used for commercial nuclear power production. Uranium ore resources appear to be assured for several decades but they are expected to become exhausted, at least for low cost uranium, in the middle (assuming high growth of nuclear energy) and towards the end of this century (assuming low growth of nuclear energy). Continued exploration efforts may further extend those uranium resources but probably at a higher cost.

More efficient use of ore can be achieved in three ways by:

- more extraction of fissile material from existing stocks of depleted uranium;
- improved fuel-efficient reactors;
- more recycling of existing discharged SNF.

An attractive additional resource of natural fertile material will be the use of ores containing thorium. There is some reactor technology know-how available how to use thorium, in particular from the US development of the FSV-HTGR and the German development of the THTR-300, which was designed in order to convert ^{232}Th into ^{233}U. Unfortunately, these developments were stopped in 1989 and 1988, respectively, mainly for political reasons.

While the fuel cycle today contributes only about 20% to the overall cost of energy, these first general remarks show already how important it is to have well adjusted nuclear fuel cycles available to make the best use of the natural resources that are available on the long range.

3.2.1. Definition of closed and open fuel cycles

Any cycle is supposed to end where it started. In the case of Nuclear Fuel Cycles (NFC) this means that the cycle ends there where the nuclear fuel raw material at the beginning was taken from: in the ground.

In principle two types of NFC could be considered:

- a closed nuclear fuel cycle;
- an open nuclear fuel cycle, also called once-through cycles.

A closed fuel cycle means that — theoretically — all fissile material, left over or produced during reactor operation, will be returned to the reactor for further conversion of energy. Back

to the ground goes only the waste with no more potential for energy conversion, i.e. the fission products and any other material that is not fissile within the used technology.

An open or once-through NFC sends all spent nuclear fuel (SNF) back to the ground, after being used in a reactor only once and regardless how much fissile (and fertile) material is still in this fuel.

Figure 3.1 shows the scheme of a closed NFC using uranium and plutonium as the fuel:

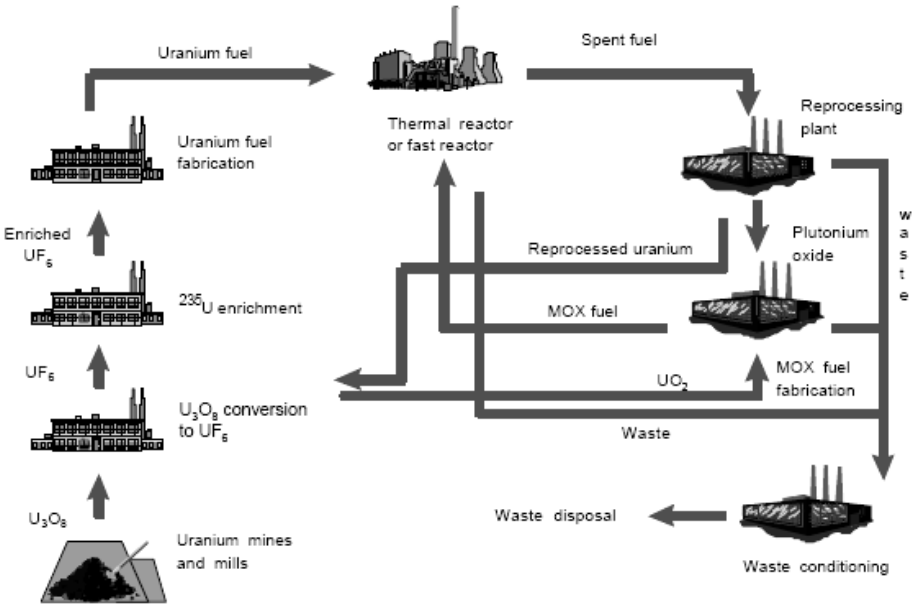


FIG. 3.1. Scheme of a closed U/Pu NFC.

The next Figure 3.2 shows an open NFC using only uranium as the fuel.

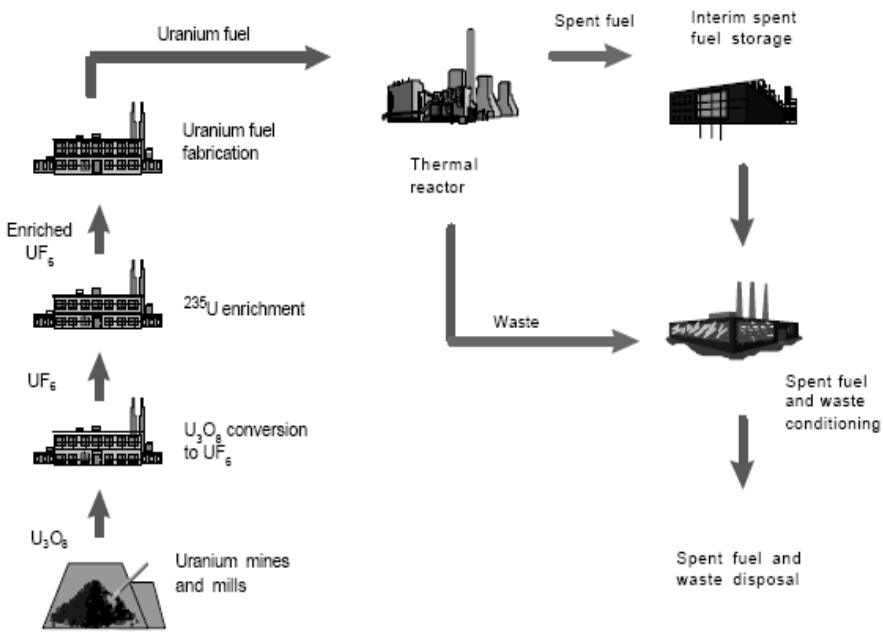


FIG. 3.2. Scheme of an open uranium NFC.

However, the historical development of nuclear energy has shown that a more differentiated view is necessary to reflect the realities of nuclear fuel cycles. Following the NFC studies

within the Generation IV activities (Ref. [38] and Section 2.6.1), four different NFCs could be defined (Fig. 3.3).

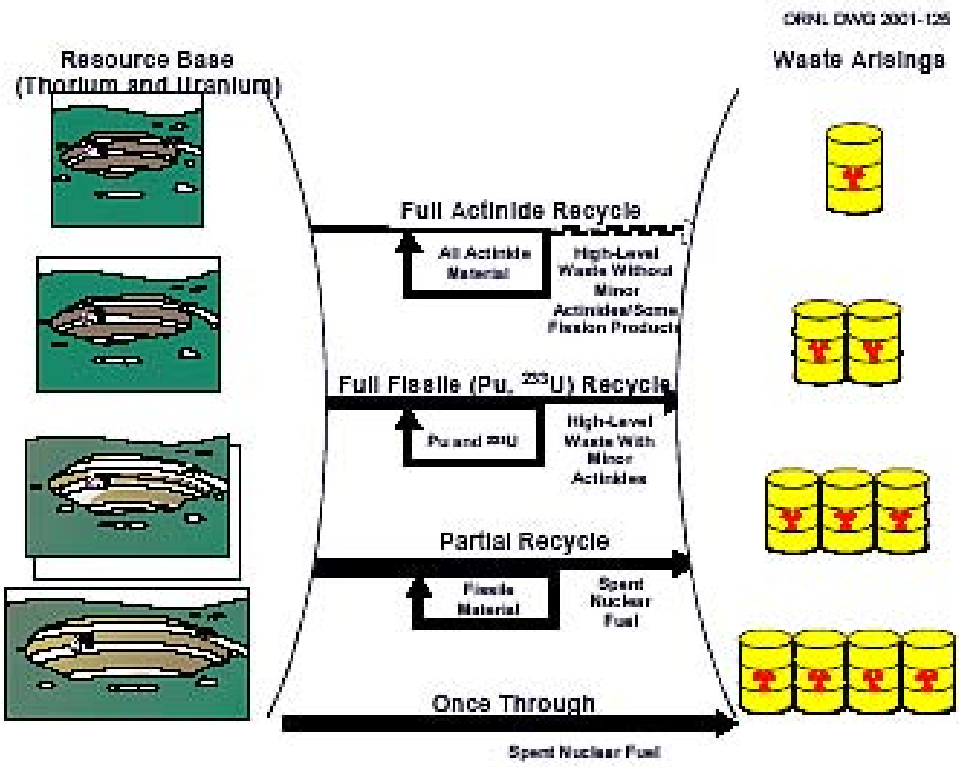


FIG. 3.3. Generation IV: Alternative nuclear fuel cycles.

In the following list, the four different NFC variants above (Fig. 3.3) are explained in more detail.

(1) Once-through NFC: The fuel is fabricated from uranium (and thorium), irradiated in the core, discharged as spent nuclear fuel (SNF) and stored for a period to allow for reduction of decay heat, then directly disposed of as a high level waste (HLW). Light water reactors (LWRs) in the United States of America currently use such a fuel cycle with uranium.

(2) Partial recycle NFC: Some fraction of the SNF is processed, and some fraction of the (actinide) fissile material is recovered by recycling, and new fuel is fabricated with this material. The fuel is returned to the reactor one time or several times (generally not exceeding three) to extract additional energy and the resulting SNF is then disposed as waste. One example is the French recycling system, in which (i) low enriched uranium SNF is recycled back to thermal reactors as MOX fuel, and (ii) the resultant MOX SNF may be directly disposed of. A second example is the proposed DUPIC fuel cycle, which converts LWR SNF into new CANDU fuel with direct disposal of this CANDU SNF.

(3) Full fissile recycle NFC: All SNF is processed for recovery and recycle of fissile material (^{235}U, plutonium and/or ^{233}U). The SNF is repeatedly processed and recycled to fully consume all the fissile material through multiple usages in a reactor. Minor actinides and fission products are separated and sent to the waste stream from the processing operation. An example of this is the traditional liquid metal fast breeder reactor (LMFBR) fuel cycle. However, this cycle has not yet been implemented at an industrial scale.

(4) Actinide and long lived fission products recycle NFC: All SNF is processed, and all actinides are multiply recycled to fully consume the fissionable material. One (or more)

fission product(s) (e.g. ^{99}Tc and ^{129}I) may be recycled as well. An example of such a fuel cycle is a combination of LWRs, liquid metal fast reactors (LMRs) and molten salt reactors. In such a system the LWRs produce power, the LMRs produce power and manufacture excess fissile material from fertile ^{238}U (or ^{232}Th) to fuel the LWRs, and the molten salt reactor is used to destroy higher actinides that would otherwise be sent to the final repository. This kind of an NFC is currently under development in some countries at laboratory scale.

Figure 3.3 also shows qualitatively how the needed resources and the final waste decrease from alternative number 1 to number 4, thus contributing more and more to the sustainability of the variant.

3.2.2. Status of worldwide use of open and closed fuel cycle

According to an evaluation of the IAEA published in 2001 (Ref. [49]), ten countries were practicing the closed NFC and six countries have decided to practice the open NFC. Ten countries have not yet decided and eight countries did not provide relevant information. Practically all fuel used in these two versions of NFC is U (Pu)-based.

That means that around 50% of the worldwide installed commercial nuclear power-plant (gross) capacity of about 370 GW(e) is at least partially operating in a closed NFC with U and as applicable Pu. The other 50% commercial reactor capacity works with an open fuel cycle (and U-based fuel). About 35% operate with a political decision for the open NFC, the rest of 15% works de facto the same way, but without a clear decision.

Table 3.2 shows the current situation as of 2007. Compared to 2001, Belgium, Germany and the Netherlands have changed to an open fuel cycle, and the USA has announced to change to a closed fuel cycle.

For both parties, there would be more or less mature technology available for direct disposal of HLW. But, currently, for none of the 100% commercial reactor's NFC a fully proven and practiced HLW final disposal facility exists, not for the products after reprocessing and not for spent fuel disposal. This is a big disadvantage for nuclear energy and of course, unfortunately a strong argument for those who are opposing nuclear energy.

Up to now, only the U oxide-based NFC variants have reached a high enough industrial maturity and experience base for the closed variant of the NFC. This of course leads to consequences for the assessment of the future perspectives of coming NFC solutions.

TABLE 3.2. CLOSED AND OPEN NUCLEAR FUEL CYCLES APPLIED BY COUNTRY (REPRODUCED FROM REF. [49] AND UPDATED IN 2007)

Country	Closed NFC	Open NFC	Not yet decided	Remarks
Argentina			x	
Armenia			x	
Belgium[a]	x [a]	x [a]		
Brazil			x	
Bulgaria			x	
Canada		x		
China	x			
Czech Republic				No information

Note: [a] Belgium changed to an open fuel cycle after 1998.

TABLE 3.2. CLOSED AND OPEN NUCLEAR FUEL CYCLES APPLIED BY COUNTRY (REPRODUCED FROM REF. [49] AND UPDATED IN 2007) (cont.)

Country	Closed NFC	Open NFC	Not yet decided	Remarks
Finland		x		
France	x			
Germany [b]	x[b]	x[b]		
Hungary			x	
India	x			
Japan	x			
Kazakhstan				No information
Republic of Korea			x	
Lithuania				No information
Mexico				No information
Netherlands[c]	x[c]	x[c]		
Pakistan				No information
Romania				No information
Russian Federation	x			
Slovakia			x	No information
Slovenia			x	
South Africa			x	
Spain			x	
Sweden		x		
Switzerland	x			
Ukraine			x	
United Kingdom	x			
USA[d]	x[d]	x[d]		

Notes:
[b] Germany changed to an open fuel cycle in 2005.
[c] The Netherlands changed to an open fuel cycle in the 1990s.
[d] The USA announced change to closed fuel cycle in 2006.

3.2.3. Current status and strategies of NFC in specific countries

Today NFC services are no longer being exclusively performed in OECD countries. Twelve countries outside the OECD have built fuel cycle facilities, and many steps of the total fuel cycle can be found under development or in operation in one or more of these countries. In the following Sections 3.2.3.1 to 3.2.3.14 examples of national NFC practices are described as far as there are clear national political decisions and legislative conditions on the use of a closed or open NFC.

3.2.3.1. Situation in Belgium

In Belgium, there is no mining and milling, no conversion, no enrichment and no reprocessing technology deployed. Uranium (U_3O_8) comes from various sources (Australia, Canada, China, France, Niger, South Africa and the USA). Conversion to UF_6 is performed in Canada, France, the UK and the USA. Enrichment services are purchased in France, in the Russian Federation, at URENCO (a British/Dutch/German-joint venture) and in the USA.

The fuel for all PWR is now provided by AREVA NP (formerly called Framatome-ANP with fabrication in France, Germany and the USA) and Westinghouse (with fabrication in Spain, Sweden and the USA). 7 PWRs (together gross 6 GW(e)) are in operation.

Reprocessing of Belgian SNF has been performed in France till 1998. Thereafter, wet storage of SNF is being used at the Tihange nuclear power plant and dry storage at the Doel nuclear power plant.

For the low level waste, local partnerships in some municipalities are preparing integrated projects, incorporating the disposal facility in a broader development of the region. For the long lived, medium and high level waste, research is going on in the underground research laboratory in the Boom clay, which has been recently extended. In the extension, large scale experiments (heater and plug tests) are foreseen. A choice has also been made on the disposal concept. The super container will be used in the future as the reference concept.

Belgium is involved in a number of research projects on partitioning and transmutation in the European framework programmes. In particular, it is studying a multipurpose accelerator driven system, which will allow many applications, including a study of the transmutation of minor actinides and long lived fission products (PUMA, MARHA[10]).

There are no definite plans for introduction of fast breeder reactors or multiple recycling of fissile material in Belgium. Only mono-recycling of plutonium is performed in two PWRs.

3.2.3.2. Situation in Canada

Canada retained its position as world leader in uranium production in 2002 with output totalling 11607 t·U (tonnes of uranium metal), down slightly from the 2001 total, mainly due to reduced Rabbit Lake mine output. As of 1 January 2005, Canada's recoverable uranium resources amounted to 444000 t·U, down slightly from the 2002 total of 452000 t·U due to extraction and ongoing deposit appraisal.

With over 85% of the resource base categorized as low cost, Canada is well positioned to continue its leadership in uranium production. Canadian uranium production is supposed to remain constant in 2005 at about 11600 t·U.

Milling is performed in Key Lake and Rabbit Lake. Conversion is performed in the Blind River plant, and the Port Hope plant, both in Ontario. There is no enrichment and no reprocessing in Canada.

GE Canada operates a fabrication plant for PHWR fuel assembly (CANDU-type) at Peterborough, Ontario and a pellet fabrication plant in Toronto. Zircatec Precision Industries operates a plant for PHWR fuel assembly (18 CANDU reactors with together gross 13 GW(e)) are in operation.

Spent nuclear fuel (SNF) is stored on-site in pools or dry silos. For long term management of this waste, Atomic Energy of Canada Ltd (AECL) has developed a deep geological disposal concept, which is not yet publicly accepted. The year 2015 is considered an achievable target date for bringing a long term waste management facility into service.

In accordance with their strategy to use an open fuel cycle, there are no plans for introducing fast reactors into the NFC.

[10] http://www.sckcen.be/marha/

3.2.3.3. Situation in China

In China, mining and milling is being performed in several plants (Fuzhou centre, Chongyi centre, Yining centre, Lantian centre and Benxi centre).

A UF_6 conversion facility is operating near Lanzhou.

There are two enrichment plants; one is the Lanzhou Nuclear Fuel Complex, which originally used the diffusion and now only applies centrifugal processes; the other one is the Shanxi Nuclear Fuel Complex (Hanzhong), which also uses the centrifugal process. The total separative work capacity is reported [49] to match the demands of the nuclear power development programme (a total of 20 GW(e) nuclear capacity by 2010 and 32–36 GW(e) by 2020).

Fuel fabrication plants are in Ybin (PWR fuel, to be increased for the total fuel demand according to the Chinese nuclear power plan) and in Baotou (CANDU fuel, operating since 2002).

A civil reprocessing pilot plant with a capacity of 300 kg HM/d is under construction in Lanzhou. Also in Lanzhou a wet storage facility is under construction.

There are well defined programmes for introducing fast breeder reactors and high temperature gas cooled reactors with their corresponding fuel cycles.

3.2.3.4. Situation in Finland

In Finland, there is no mining and milling, no conversion, no enrichment and no reprocessing in operation. Uranium (U_3O_8) comes from Australia, Canada, China, the Russian Federation and the USA. Conversion services are purchased in Canada, France, the Russian Federation and the UK. Enrichment services are provided from the Russian Federation and URENCO (a British/Dutch/German-joint venture).

There is no domestic fuel fabrication. All fuel assemblies are imported from the international fuel vendors. (2 WWER plants with together gross 1020 MW(e) and 2 BWR with together gross 1740 MW(e) are in operation; also a new PWR plant (EPR) is under construction).

Formerly, spent fuel from Loviisa NPP was sent to the Russian Federation. Now, an interim spent fuel storage facility is in operation at the Loviisa NPP and an extension is underway. Also, at the Olkiluoto NPP a wet storage facility for spent fuel is in operation. A project for the final disposal is in a planning stage. Recently, an underground rock facility for final storage of spent fuel is under construction.

3.2.3.5. Situation in France

In France, no more domestic mining and milling is in operation. However, the French company Cogema (now part of AREVA FC) is running several operations abroad, i.e. in Niger, Canada and the USA.

Conversion is performed in France, the USA and the Russian Federation.

Enrichment services are coming from France (Tricastin), the Russian Federation, URENCO (a British/Dutch/German-join venture) and the USA.

Fuel (for 58 PWR with gross 65 GW(e)) is now mainly provided by AREVA NP (with fabrication in France, Germany, and the USA) and to a smaller extent by Westinghouse (with fabrication in Spain, Sweden, and the USA).

The French strategy regarding the fuel cycle is illustrated in the following Fig. 3.4.

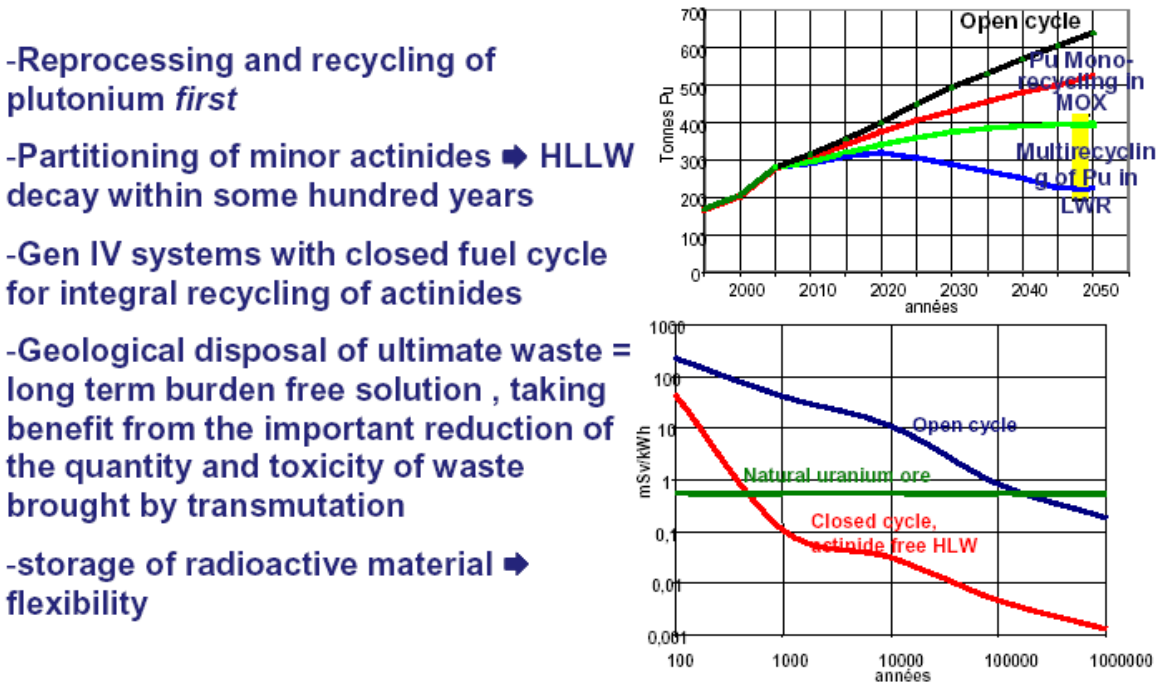


FIG. 3.4. France's fuel cycle strategy.

French PWR fuel is sent to the French La Hague pools for cooling before undergoing reprocessing at the Cogema UP2 plant. Also large quantities of foreign fuel are reprocessed at Cogema's UP3 plant located in La Hague.

Recovered plutonium is reused in the form of MOX fuel for which a specific fuel fabrication plant called MELOX is operating in Marcoule. Part of the recovered uranium (or reprocessed uranium RepU) is re-enriched and used for the fabrication of fresh uranium fuel feeding three reactors [50]. High level wastes (HLW) are conditioned and stored for future transfer to the French Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA) or — in case of foreign fuel — back to the foreign customers. Low level wastes are already transported to the ANDRA site.

3.2.3.6. Situation in Germany

In Germany, there is no mining and milling and no conversion. Uranium (U_3O_8) comes from practically all international suppliers (using material from Australia, Canada, China, France, Gabon, Niger, Portugal, South Africa and the USA). Conversion to UF_6 is performed in Canada, France, the UK and the USA.

However, Germany is one of the three participants of the enrichment company URENCO. Enrichment services for German reactors are purchased in France, in the Russian Federation, at URENCO (a British/Dutch/German-joint venture) and in the USA.

Fuel (for BWR and PWR) is now mainly provided by AREVA NP, and to a smaller extend by Westinghouse.

There are no domestic reprocessing activities; however reprocessing was performed for part of the spent fuel of German reactors in France (La Hague) and in the UK (Sellafield). Reprocessed Pu (MOX) and U (RepU) is recycled in German reactors. Spent fuel not shipped for reprocessing is stored partly in central interim storage facilities and partly on reactor sites. High level waste from reprocessing comes back to a German central storage facility.

The German nuclear phase out policy forbids reprocessing of spent fuel since 2005, i.e. Germany applies an open fuel cycle strategy currently.

3.2.2.7. Situation in India

In India, four underground uranium mines (Jaduguda, Bhatin, Narwapahar and Turamdih) are in operation. In addition, in three plants uranium mineral concentrates are recovered from copper mill tails.

All the uranium ore together with uranium mineral concentrates are processed in a mill at Jaduguda and converted into Magnesium-Di-Uranium (MDU) or yellow cake and sent to the Nuclear Fuel Complex at Hyderabad (NFCH) for the conversion to UO_2 and manufacturing of Zircaloy clad PHWR fuel bundles. NFCH also supplies the fuel bundles for India's BWRs (fabricated using enriched UF_6). NFCH is also responsible for production of zirconium alloy cladding tubes and fuel assembly components using zircon as starting material from the Indian rare earth plants in southern India.

India has pilot plants for manufacturing mixed uranium/plutonium oxide (MOX) and mono-carbide (MC) fuels. The MOX pilot plant has been utilized for fabrication of some 12 Zircaloy clad MOX fuel assemblies (6 × 6), which have been successfully irradiated in the two BWRs at Tarapur to burnups in the range of 20000 MW·d/t. Recently 50 MOX fuel bundles were loaded in one of the PHWRs (KAPS-1).

India developed a technology for reprocessing natural uranium oxide fuel from its CANDU type reactors using a pilot plant at Tarapur. A second plant is in service at Kalpakkam.

A pilot plant, based on THOREX process, is in operation for reprocessing irradiated metallic thorium and ThO_2 from research and power reactors respectively. The ^{233}U thus recovered has been utilized as Al-20% ^{233}U fuel for the 30 kW water cooled KAMINI research reactor for neutron radiography.

Finally, several heavy water production facilities (in Baroda, Kota, Manuguru, Talcher and Tuticorin) are in operation.

As mentioned before (Section 2.3.5) India is clearly working towards the introduction of fast breeder reactors, and advanced HWRs (with a breeding capacity) using a Th/^{233}U fuel cycle.

3.2.2.8. Situation in Japan

Japan has no indigenous resources of oil, gas and coal and is not connected to overseas pipelines and, therefore, is developing a national strategy that includes MOX utilization in LWR and FR.

There are no commercial mining and milling and also no conversion facilities. Uranium raw material comes from Australia, Canada, China, France, Niger, South Africa and the USA. Conversion to UF_6 is performed in Canada, France, South Africa, the UK and the USA.

A commercial enrichment plant is operated at Rokkasho-mura in Japan. Its capacity is to be expanded since so far about 80% of the enrichment was performed in Europe (URENCO), France and in the USA.

Most of the uranium LWR fuel is fabricated in Japan in four facilities. MOX fuel has been fabricated in three small scale plants (PFDF, PFFF, PFPF FBR) in JNC. The JNFL MOX fabrication plant for LWR fuel will be operated with a production capacity of 130 t/a in Rokkasho in 2009. In three plants Zircaloy tubes are fabricated.

Spent LWR fuel is reprocessed up till now at the small Tokai reprocessing plant (TRP) and foreign plants (UK). Until May 2007, TRP has processed 1029 t of LWR fuel, 82 t of UO2 FUGEN fuel and about 30 t of FUGEN MOX fuel [51]. The commercial Rokkasho reprocessing plant (RRP) with a planned capacity of 800 t/a has finished non-radioactive test operation in January 2006. Testing with radioactive material has been started in March 2006 [52]. Recovered plutonium is to be reused in LWR and FR fuel; no decision for use of uranium (RepU) has been made.

Nuclear Waste Management Organization of Japan (NUMO) in will start the disposal of high level radioactive waste about the mid-2030s.

Japan is aiming to continue the closed fuel cycle with mono-recycling of plutonium in thermal reactors and multiple recycling in fast reactors.

3.2.3.9. Situation in the Republic of Korea

In the Republic of Korea there is no domestic mining and milling, no commercial conversion (just one pilot plant), and no enrichment plant. Uranium (U_3O_8) comes from Australia, Canada, China, France, Namibia, Niger, the Russian Federation, South Africa and the USA. Conversion services are purchased in Canada, France, the Russian Federation and the UK. Enrichment services are provided from France, the Russian Federation, URENCO (a British/Dutch/German-joint venture) and the USA.

The Republic of Korea relies solely on overseas U yellow cake. It is expected that worldwide uranium supply will be more than sufficient for the time being, and the price will remain stable. Because the share of nuclear power in the world energy supply will steadily increase, there may be, however, a possibility to see a rise of the uranium price in a long term perspective. Therefore, in order to ensure long term undisturbed supply, it will be prudent to diversify the supplying countries. Stable supply of yellow cake may be assured through both long term purchase contracts and development of overseas uranium mines. Purchase from spot markets will remain as a supplementary option to diversify supply sources and improve contractual terms. In order to be prepared against unexpected energy crisis or emergency, potential domestic uranium deposits will be explored as well.

Regarding enrichment, for the Republic of Korea it is necessary to rely on foreign services. Services will be diversified to several countries to stabilize the availability.

Korean nuclear industry has achieved technical self-reliance to a great extent in the field of nuclear fuel design and manufacturing. At present, design and manufacturing technology to supply nuclear fuels to the operating CANDUs and PWRs is available in the Republic of Korea from the Korea Nuclear Fuel Company Ltd (KNFC). KNFC has operated the ammonium uranyl carbonate re-conversion process for scrap recycle and the dry re-conversion process (capacity 400 t·U/a) from 1998, as well as converting imported enriched uranium from UF_6 to UO_2.

But, as the number of plants increases, manufacturing capacity will be added in time to meet the demand. Regarding fuel design and manufacturing, the capacity surplus, if any, may be used to provide services to foreign countries.

Currently, all spent fuels are stored at each plant.

The Korean nuclear fuel cycle will be developed for better utilization of uranium resources, minimization of radioactive wastes, and energy security. Efforts will be undertaken to fulfil the requirement that the nuclear fuel cycle should be in harmony with reactor strategy, to improve the overall economics and the safety of nuclear power (Fig. 3.5). Fuel cycle technology will be established based upon understanding and support from the international

community in the frame of international non-proliferation regime of NPT and the spirit of Declaration of Nuclear Non-Proliferation on the Korean Peninsula.

With the vision of better utilization of uranium resources, less generation of spent fuel, and better economics of nuclear fuel, the Republic of Korea is paying attention to the technical and commercial feasibility of the mixed oxide (MOX) fuel assemblies loaded in power reactors.

If necessary, the Republic of Korea could consider the possibility of using foreign services for spent fuel reprocessing and MOX fuel fabrication. The PWR spent fuel with remaining excess reactivity can be a good fuel resource for CANDU reactors. As mentioned before, the Republic of Korea has both PWRs and CANDU reactors in operation. Thus, the Republic of Korea can anticipate synergistic benefits of saving CANDU fuel and reducing spent PWR fuel storage requirement, by connecting the fuel cycles of the two reactor types (DUPIC, see Section 3.4.1).

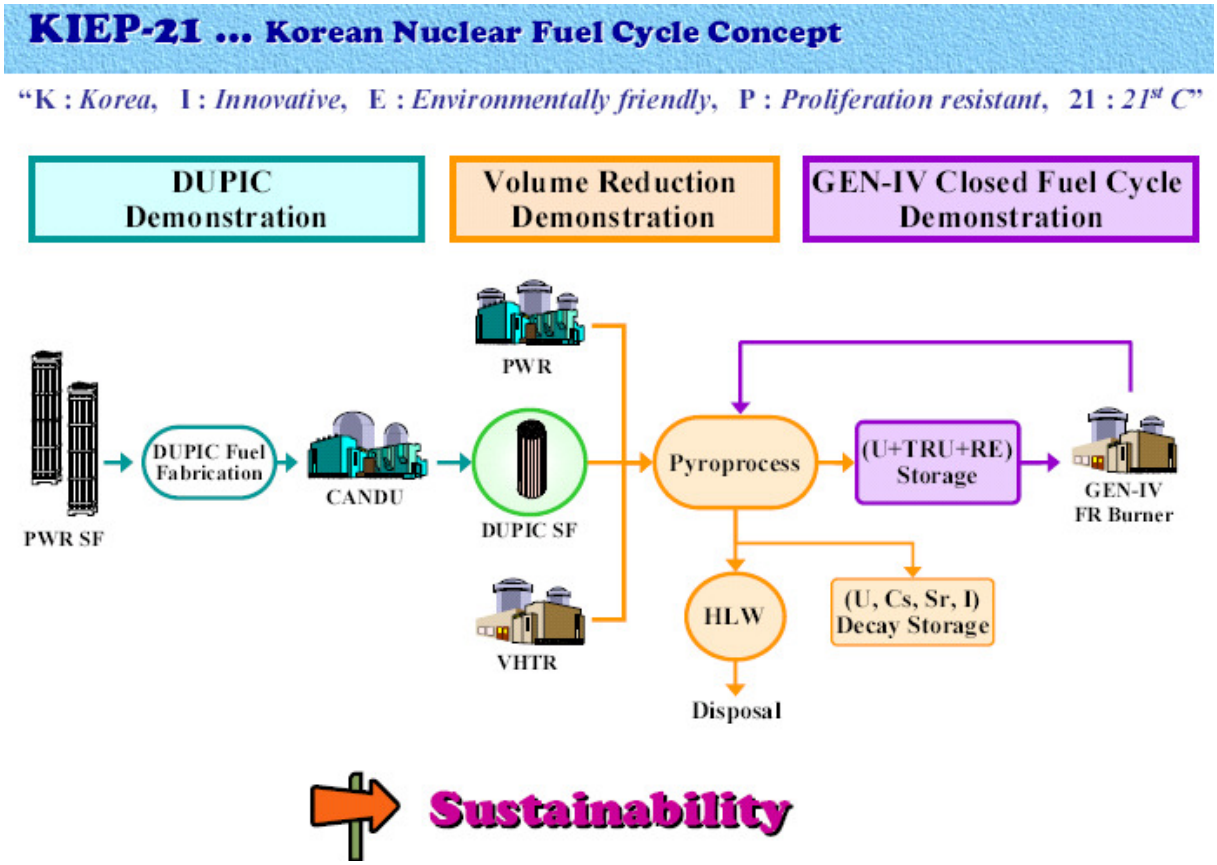


FIG. 3.5. Korean nuclear fuel cycle concept (IAEA TWG NFCO 2006).

3.2.3.10. Situation in the Netherlands

There is no mining, milling and conversion in the Netherlands. Only uranium enrichment is carried out by a company called Urenco (owned by the Netherlands, the UK and Germany), located at Marlow. The rest of the fuel cycle (i.e. the fuel elements) is purchased on the world market.

In the Netherlands, a closed NFC was practiced with two plants (a BWR and a PWR) till the 1990s. Spent fuel of the two plants was partly reprocessed in France, and the plutonium recycled is both stored in the country and sold to other foreign utilities for production of MOX fuel.

A private company, called NV Cora, is in charge of radioactive waste management. It is owned by the major nuclear waste producers in the Netherlands, e.g. Dodewaard, Borssele, ECN. A licence for a final depository of nuclear waste has not been granted yet.

3.2.3.11. Situation in the Russian Federation

The Russian Federation has full capability in all segments of the nuclear fuel cycle. Mining and milling is performed in commercial scale (using open pit, underground and ISL extraction methods) for the domestic demand and for the export of WWER and RMBK fuel.

Commercial conversion is performed in Angarsk, Ekaterinburg and Tomsk. Excess capacities are offered commercially to foreign utilities.

Four commercial enrichment plants are operated in Sverdlovsk, Tomsk, Angarsk and Krasnoyarsk. Again excess capacities are offered commercially to foreign utilities.

Fuel fabrication is performed at Electrostal near Moscow and in Novosibirsk. The operation in Elektrostal provides fuel for WWER-440, WWER-1000, BN-350, BN-600 and RMBK reactors, and the plant in Novosibirsk for WWER-1000 units (31 reactors are in operation with gross 21,7 GW(e) capacity). A new line for pellet production at the Novosibirsk plant started operation in 2000[11]. Zirconium production for nuclear fuel is performed at the Glazov plant. Also free fuel fabrication capacities are offered to foreign companies.

Reprocessing is done for the Russian LWRs and fast reactors, with the exception of RBMK fuel, which shall be disposed as spent fuel. At present there is one plant in PO Mayak at Ozersk (Chelyabinsk-65) operated to reprocess fuel from WWER-440 reactors, fast reactors (and the propulsion reactors of ice-breakers and submarines).

The construction of a second reprocessing plant (RT-2) at Krasnoyarsk, has been postponed indefinitely. Presently, the treatment and use of weapon plutonium has priority in the Russian Federation. Reprocessed uranium is used for RBMK fuel production. Plutonium obtained at the RT-1 in PO Mayak at Ozersk (Chelyabinsk-65) is temporarily stored on-site in dioxide form.

Several wet AFR (away from reactor) fuel storage facilities are operated in PO Mayak at Ozersk (Chelyabinsk-65) and at RT-1 and RT-2, and in GChK at Zheleznogorsk (Krasnoyarsk-26), at several nuclear power plants. However, in future the dry storage technology will be used.

The nuclear fuel cycle strategy is to recycle plutonium (currently produced in thermal reactors) only in fast breeder reactors to be built in the future.

3.2.3.12. Situation in Switzerland

In Switzerland, there is no mining and milling, no conversion and no enrichment plant operating. And there is also no domestic fuel fabrication. U-oxide and MOX fuel is imported from western fuel suppliers (2 BWRs with together gross 1572 MW(e) and 3 PWRs with together gross 1780 MW(e)) are in operation.

Spent fuel is reprocessed in the UK and in France. Mono-recycling of reprocessed plutonium is performed. There are no plans for the introduction of fast breeders into the fuel cycle.

Dry spent fuel interim storage facilities are under construction (ZWILAG).

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[11] Originally, in Ust Kamenogorsk (Kazakhstan) UO_2 pellets were fabricated.

3.2.3.13. Situation in the UK

There is no mining and milling of uranium in the UK. All other parts of the NFC are performed in the UK for domestic and foreign reactors.

Enrichment is performed by URENCO at Capenhorst. Fuel (uranium and MOX) is fabricated at the BNFL's Springfield site. Reprocessing is performed in the THORP facility at Sellafield. High level waste is to taken care of by an organization called NIREX.

Currently, there are no definitive plans for recycling of spent fuel in thermal or fast reactors.

3.2.3.14. Situation in the USA

In the USA, 15 uranium mines and 7 milling facilities are in operation (2004), but only one commercial UF_6 conversion plant is operating. Two commercial enrichment plants are operating at Paducah, Kentucky, and Portsmouth, Ohio.

Fuel (BWR and PWR) is now provided by AREVA NP, Global Nuclear Fuel (former GE nuclear fuel) and by Westinghouse.

About 95% of the spent fuel is stored on-site in reactor pools. Storage in many pools has been increased over the years with high density racks. Some US utilities are using additional dry storage systems at their reactor sites.

Licensing of a geologic repository is underway at Yucca Mountain, Nevada, launched in 2004, with the aim of potential retrievability of spent fuel for a period of up to 300 years. For the near term, however, the trend toward increased use of onsite dry storage is expected to continue.

As shortly laid out in Section 2.6.2, the US government has recently (2006) launched an international initiative called GNEP that could have an impact on the back end of the US nuclear fuel cycle by reintroducing reprocessing.

3.3. Trends of nuclear fuel cycle technologies

3.3.1. Introduction

A global survey on NFC trends considering economic, environmental and social aspects was performed by the Nuclear Energy Agency (NEA) within OECD [53], discussing a broad spectrum of NFC variants. According to this study:

- The short term developments will concentrate on efforts to reduce the cost of existing NFC focusing on evolutionary changes, i.e. no investment in radically new technologies.
- In a medium time range development efforts will mainly aim at solutions improving the NFC back end, in particular the management of the long living transuranics in spent fuel.
- Other possible developments are expected that may adapt the NFC to the re-use and to new reactor systems such as the high temperature gas cooled reactor, that allow non-electrical applications such as the generation of process heat.
- Only after the next 20 years development of more advanced NFC technologies may be envisaged, aiming at improved back end solutions in three directions:
    - Introducing additional waste management options, such as partitioning and transmutation, i.e. closing the NFC not only for Pu but also for the minor

actinides; these strategies will first still rely on LWR technology, and later also include fast reactor systems.

- Going to fully integrated cycles, such as 100% fast reactors cycles or even molten salt fuel cycles.
- Changing from the U/Pu cycle to a Th/^{233}U cycle, where the impact of long lived radioisotopes is reduced; those concepts could use modified LWR, HTR, or MSR.

While any requirements for economics, safety and non-proliferation can be solved primarily by technological approaches, requirements for sustainability and environment fundamentally include the issue of available resources and their adequate use. Therefore, before starting considerations on promising further developed or new technologies over medium (<25 years) to long (25–50 years) ranges, an analysis of the present situation and future perspectives of nuclear energy is necessary that evaluates the availability and adequate use of the resources needed for a long range development of nuclear power production.

National and international institutions considering the interaction of available nuclear resources and potential nuclear technologies have performed many studies (an example is presented in Section 3.3.3). Most of these studies come to the conclusion that with the existing U/Pu thermal fuel cycle the today known and economically acceptable U resources will be exhausted before the end of this century, if the present level of nuclear energy production is about maintained. This time horizon may be longer with unexpected new U resources, but it will be also shorter, if the volume of nuclear energy production increases during the next decades.

Therefore, there is general agreement that a reasonable contribution of nuclear energy to satisfy the overall long range worldwide energy demand can only be achieved by using nuclear breeder technologies within closed nuclear fuel cycles.

An example of such a system is the self consistent nuclear energy system SCNES [54], a Japanese concept that is sustainable in the long term with a high efficiency regarding use of resources, increased proliferation resistance and minimization of the impact on the environment.

3.3.2. Technical potential of NFC

Political trends appear to change slowly. Sustainability of the energy resources and global climate change issues are becoming more and more important worldwide. In the USA, energy independence is again becoming a political issue due to changes of strategic targets and again increasing uncertainties of oil supplies. Slowly also the public acceptance of nuclear power appears to change, in particular today in the USA.

On the one side, an advantage of nuclear energy becomes more evident, i.e. nuclear raw materials have a much smaller energy specific volume as compared to any other energy relevant raw materials. Therefore nuclear raw materials can be handled and stored much easier than other energy sources. This can be a big help to ensure a reliable national energy supply.

On the other side, the resources of nuclear energy raw materials are limited too, but with a chance to increase the effectiveness of their use dramatically. The already visible technical possibilities of much improved use of these resources have to be understood as a big chance for a much less politically sensitive energy supply worldwide.

The technical potentials of NFC are:

- achieving higher burnup in once-through cycles (with higher enrichment), but burnup has only very limited influence on natural U-consumption;

- decreasing the lower tails in enrichment processes;
- introducing advanced reactors with higher thermal efficiencies;
- extending the thermal uranium and plutonium recycling, which includes of course a broader deployment of reprocessing;
- starting reactor and NFC technologies to use thorium as a fertile feed material.

The introduction of advanced NFC technologies for better resource effectiveness has to go along towards minimizing their environmental impacts. This holds particularly for mining and milling technologies of the front end of the NFC. At the back end the minimization of spent fuel and wastes is a big challenge for the future NFC technologies. As already discussed in Section 3.2, the more closed a NFC were, the more automatically both targets would be met together with better use of the ore resources.

In principle transmutation of long lived fission products in symbiotic fuel cycles will be able to reduce especially the burden on future generations to take care of safe final depositories for extremely long time. Various concepts are proposed with enough excess neutrons available in critical or subcritical systems, and some of them are presently studied extensively.

There remain two targets to be mentioned for the development and deployment of advanced NFC technologies. One target is a political one: to provide NFC technologies with improved proliferation resistance. There are concepts available also for this target [55]. The other target is mainly an economical one: so far during the five decades of development of the peaceful use of nuclear energy almost only the production of electricity was promoted. This left a huge market out of the perspectives for nuclear energy application: the production of high temperature heat for industrial processes. Of course one of the reasons was that for those applications the nuclear systems have to be localized very close to the industrial facility potentially using this process heat. Therefore, this target is strongly linked to the possibility to further increase the safety of the energy producing nuclear system. The high temperature gas cooled reactor theoretically has the potential for this demand. However, this reactor type still requires the development of a NFC technology handling efficiently the back end, i.e. the processing of the special U oxide- or Th oxide or U/Th–carbide-based fuel (e.g. the coated particles fuel). Of course, a once-through cycle would be technical feasible also for those fuel types; however, that would limit the potential of the high temperature gas cooled reactor in a very unattractive way and would make the use of thorium in the fuel non-effective.

3.4. NFC technologies with potential industrial deployment in <25 years

From what was shown in the previous section it is clear that, from the technical point of view, any improvement of reactor technology or fuel cycle components that are based on the long and broad experience of the U–(Pu)-based fuel will have a preferred starting position for any NFC using advanced nuclear reactor solutions deploying this type of fuel, like the advanced CANDU reactor, a supercritical water cooled reactor system (SCWR), or any fast reactor system staying with U–(Pu) oxide.

3.4.1. The nuclear fuel cycle DUPIC

One advanced NFC system of particular interest for near term realization is the direct use of spent PWR fuel in CANDU reactors, or DUPIC fuel cycle. It promises recycle without reprocessing [56]. This fuel cycle concept involves recycling spent PWR fuel into CANDU fuel for a single pass through a CANDU reactor.

The DUPIC fuel cycle employs thermal oxidation and reduction processes to mechanically break down the spent PWR fuel into a powder. In the process, most of the volatile fission

products (particularly cesium) are driven off and captured for disposal. The resulting powder is then pressed and sintered into new pellets and assembled into fuel bundles for a CANDU reactor.

There are obvious advantages of this NFC concept:

- proliferation resistance, because:
  - no production of an un-irradiated direct weapon use material;
  - highly radioactive material throughout the process;
  - the process is incapable of producing separated Pu and cannot be readily modified for it;
  - the characteristics of materials produced during the DUPIC fuel cycle are unusable for nuclear explosives.
- minimization of process waste by applying a dry thermal/mechanical process; reducing the amount of spent fuel generation from CANDU reactors by a factor of 2.

However, there are also drawbacks:

- Remote fuel fabrication technology.
- Remote fresh fuel handling in CANDU plant.
- According to the GEN IV Fuel Cycle Assessment Report it may take up to some time between the years 2010 and 2100, that CANDU reactors are built at a sufficient rate (for example, for achieving the ratio of 2 PWRs to 1 CANDU reactor) so that almost all LWR spent fuel, including PWR spent fuel, could be mono-recycled through CANDU reactors, i.e. all LWR SF is mono-recycled; none goes directly to waste.
- Also according to this GEN IV study the over all cost of this fuel cycle may be not increased, but also not reduced significantly.
- The effect on saving ore is assumed to be low, compared to a standard LWR open fuel cycle.

The DUPIC NFC development is an international programme with the participation of Korea Atomic Energy Research Institute, the Atomic Energy of Canada Limited, and the US Department of State. It is performed within participation of the IAEA Safeguards Department.

The R&D programme for DUPIC started in 1991 with a feasibility study; in 1994, the technical development began with first irradiation tests in the NRU from 1999 to 2002; a second irradiation phase in Hanaro in the Republic of Korea and in the NRU has started in 2000. The R&D programme is planned to end in 2007.

Strategy for DUPIC technology in the Republic of Korea

In order to maintain sustainable growth of nuclear energy, it is necessary to develop a fuel cycle technology, which reuses spent fuel in an economic and proliferation resistant way. Taking advantage of Korean nuclear power reactor strategy of having both PWR and CANDU for electricity generation in the Republic of Korea and potentially an attractive energy resource for recycling of the spent PWR fuel in a CANDU reactor because CANDU is the most neutron-efficient reactor available commercially, the concept of DUPIC (Fig. 3.6, direct use of spent PWR fuel in CANDU reactors) has been suggested as one of the solutions of spent PWR fuel management.

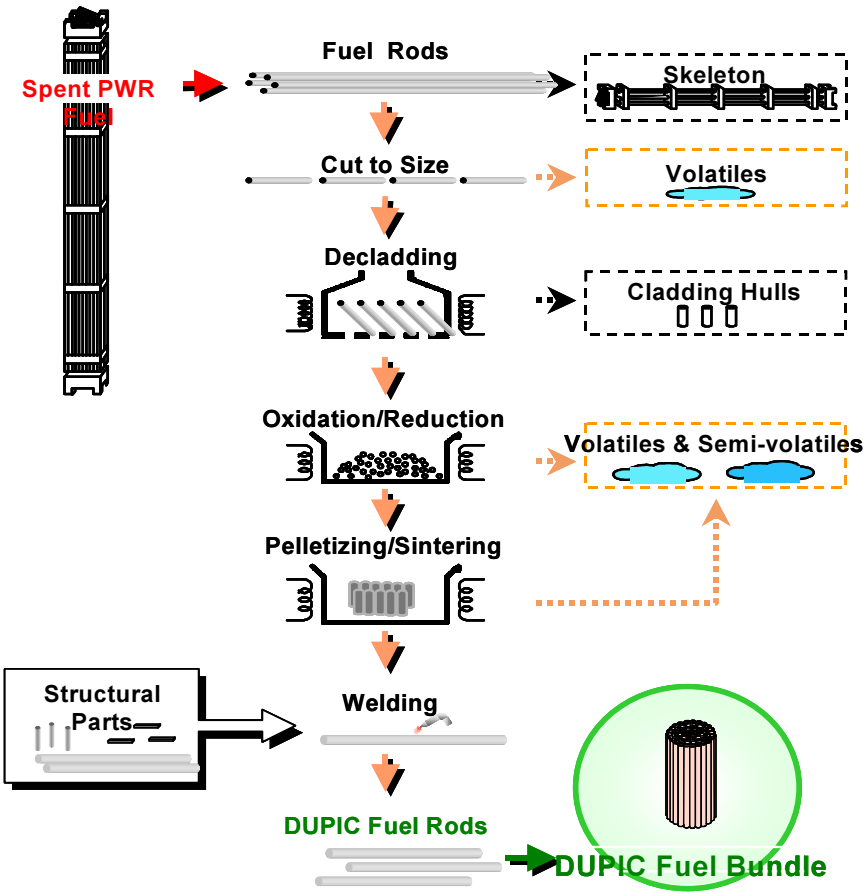

FIG. 3.6. Schematic drawing of DUPIC fuel fabrication process.

Figure 3.7 shows the long term strategy of the Republic of Korea regarding the NFC for existing PWR, CANDU, and planned FR and ADS facilities.

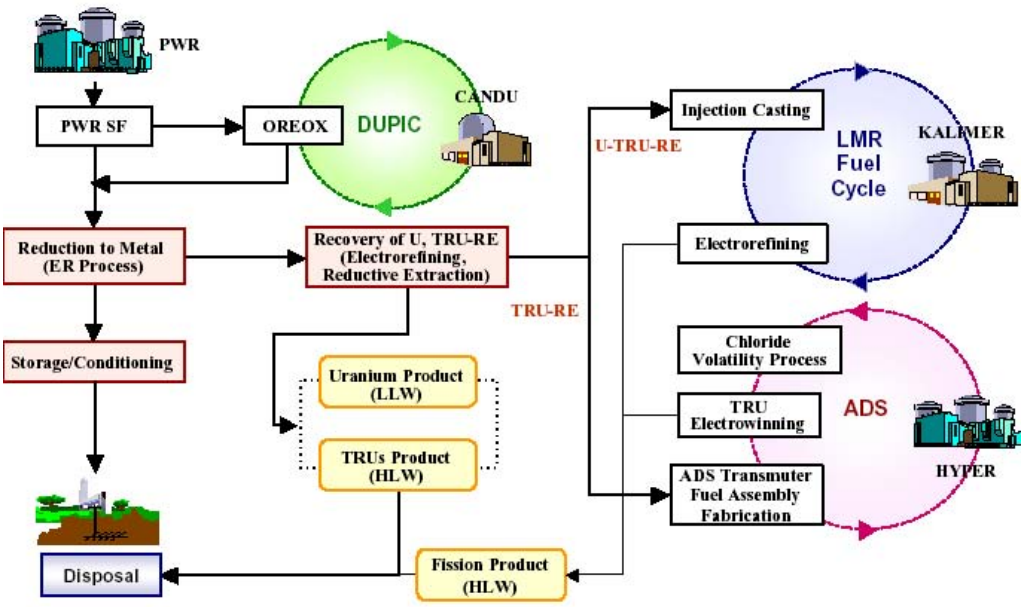

FIG. 3.7. Korean advanced fuel cycle schemes.

As illustrated above applying a DUPIC fuel cycle (Figs 3.6 and 3.7) in the Republic of Korea, would provide multiple benefits, and the main process (OREOX) of DUPIC fuel cycle is oxidation and reduction for re-fabricating a spent PWR fuel into a CANDU fuel, which is a

dry process without any separation of fissile materials (Fig. 3.8). It releases only volatile fission products, and it is thus highly proliferation resistant.

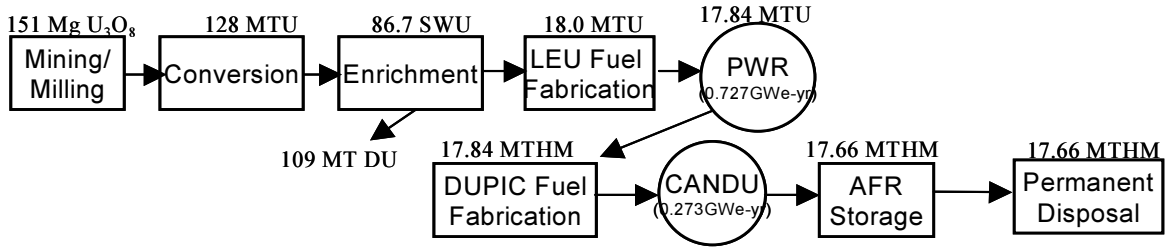

FIG. 3.8. Schematic mass flow sheet of DUPIC fuel cycle.

The synergistic effects (Fig. 3.9), which can be expected from the DUPIC linkage between LWR and HWR, in comparison with the once-through cycle, can be summarized as follows:

- Increasing proliferation resistance of the NFC: High radiation of fresh DUPIC fuel acts as a barrier against proliferation.
- Removal of spent LWR fuel: Spent LWR fuel is transformed into DUPIC fuel for burning again in the HWR; hence there can be a reduction in the repository requirements for direct disposal of LWR spent fuel.
- Saving of natural uranium and enrichment: Since the DUPIC fuel would replace natural uranium (or other enrichment) fuel for the HWR, significant resource saving can be expected in the corresponding LWR-DUPIC cycle. The environmental impact of mining the equivalent amount of uranium and its enrichment would also be avoided.
- Reduction of spent fuel arising from HWR: Since the DUPIC fuel can increase the HWR burnup (as a result of the two-fold increase in fissile content, compared to natural uranium), a further reduction in spent fuel arising from HWR can be expected using the synergism of the DUPIC fuel cycle.
- Disposal impact: The Korean ongoing DUPIC programme has recently revealed one interesting benefit pertaining to the geological disposal of spent DUPIC fuel. The DUPIC processing and subsequent irradiation in a CANDU reactor, results in a decay heat curve with time for the spent DUPIC fuel which is very similar to that of the original spent PWR fuel, before recycling. Hence, significant additional energy can be derived from the spent PWR fuel, without incurring an additional burden in decay heat.

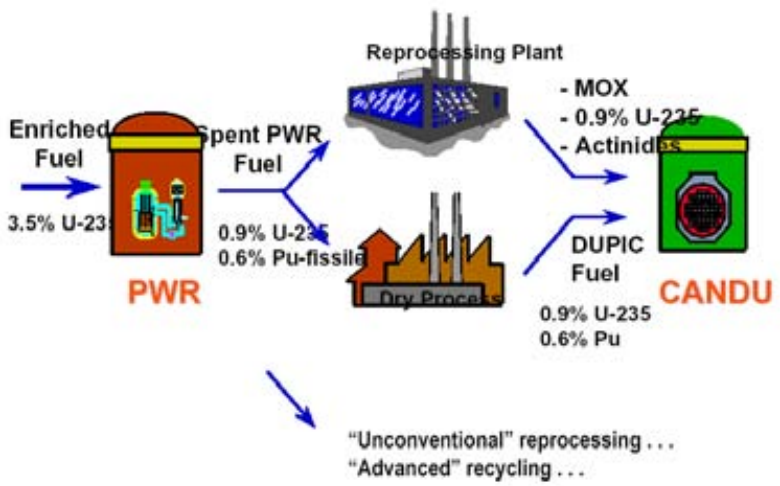

FIG. 3.9. CANDU/PWR synergism in the Republic of Korea.

3.4.2. Near term thorium fuel cycles

In the IAEA reports [57, 58] an overview is given on international and national future Th-based NFC in comparison with other NFC. In the following section a short summary of the results of these two reports is provided.

3.4.2.1. Short history of thorium fuel

Fuel that contains thorium has been used in the past for LWRs as well as for HTGRs and MSRs, but always for demonstration purposes only (Refs [59, 60]). Presently, thorium fuel is not fabricated and not used for commercial nuclear power production. Over 50 tonnes of thorium fuel in ceramic form, clad in Zircaloy, was manufactured for the Shippingport LWBR core (Section 2.1).

Significant experience was accumulated in the USA with coated particle thorium fuel in the Fort St. Vrain high temperature gas cooled reactor. Two thousand four hundred forty-eight hexagonal graphite fuel elements using 26000 kg of fissile and fertile material in TRISO-coated fuel particles were produced (this included almost 25000 kg of thorium). It was irradiated at temperatures greater than 1300ºC to a maximum burnup in the fissile material of 16% fissions of initial metal atoms, i.e. approximately 170000 MW·d/t_{HM}. Thorium-based fuel for PWRs was investigated at the Shippingport reactor in the USA using both ^{235}U and plutonium as the initial fissile material. It was concluded that thorium would not significantly affect operating strategies or core margins. The light water breeder reactor (LWBR) concept was also successfully tested in this reactor from 1977 to 1982 with thorium and ^{233}U fuel with Zircaloy cladding using the seed/blanket concept.

In the UK, Thorium fuel elements with a 10:1 Th/U (HEU) ratio were irradiated in the gas cooled Dragon reactor (20 MW(th)) at Winfrith for 741 full power days, from 1964 to 1973. The Th/U fuel was used to breed and feed, so that the ^{233}U formed replaced the ^{235}U fissioned at about the same rate, and fuel could be left in the reactor for about six years.

In Germany, the THTR reactor was built as a demonstration plant for the use of thorium fuel in coated particles, but it was operated only for a short time. Also the 60 MW(e) Lingen BWR (with conventional superheating) in Germany utilized Th/Pu-based fuel test elements.

Today, India is most interested in using thorium fuel and made thorium fuel fabrication and in-pile use to one of its major development targets (Section 2.3.5).

An overview on the specific features of a thorium fuel cycle and how to deploy it is presented in Ref. [61].

3.4.2.2. Resources of thorium

During at least the last 30 years there has been potential interest in utilizing thorium (^{232}Th) in nuclear fuel because it is supposed to be three times more available as a natural resource in the ground than uranium (Table 3.3).

TABLE 3.3. WORLD THORIUM RESOURCES (ECONOMICALLY EXTRACTABLE)

Country	Reserves / t
Australia	300000
India	290000
Norway	170000
USA	160000
Canada	100000
South Africa	35000
Brazil	16000
Other countries	95000
World total	1200000

Source: USA Geological Survey, Mineral Commodity Summaries, January 1999 referred by World Nuclear Association in 2003 (http://www.world-nuclear.org/info/inf62.htm)

3.4.2.3. Fabrication and recycling of thorium fuel

Fabrication of thorium bearing fuel per se does not raise any serious radiological problems. However, one major issue for a closed ^{232}Th–^{233}U fuel cycle is the presence of hard γ emitters (2.5 MeV) among the daughter products of the always present ^{232}U. Therefore, extra shielding and remote handling is needed during fabrication. Presently no commercial thorium fuel fabrication plants exist.

Most promising for the future use of thorium fuel is the coated particle concept as developed in Germany and in the USA for HTR fuel. During the development of prototype plants for the fabrication of mixed oxide fuels (Th/U or Th/Pu) during the 1960s in the United States of America it was found that the sol-gel process could be a very suitable process, although somewhat delicate to use and at that time not suitable for remote manufacturing facilities.

For the separation of ^{233}U and thorium a wet liquid-liquid extraction using the THOREX process was developed years ago and a pilot plant has been operated during many years at Oak Ridge (USA). The recycling of U/Th is somewhat more complicated than the traditional U/Pu cycle. However, a successful development to commercial maturity appears to be technically realistic. A major challenge is supposed to be the head-end processing of paocarbon/silicon-carbide-coated Th carbide or oxide fuel for the HTR.

3.4.2.4. Advantages and disadvantages of the use thorium NFC

Thorium is — like ^{238}U — a fertile but not a fissile material itself, but ^{232}Th will absorb neutrons to produce ^{233}U, which is fissile.

In one significant aspect ^{233}U is better than ^{235}U and ^{239}Pu i.e. its higher neutron yield per neutron absorbed in the thermal energy range. Therefore, together with some other fissile material (^{235}U or ^{239}Pu), a breeding cycle similar to but more efficient than one based on ^{238}U and plutonium (in slow neutron reactors) can be set up. The ^{232}Th absorbs a neutron to become ^{233}Th, which decays into protactinium-233 and then into ^{233}U. The irradiated (spent) fuel can then be unloaded from the reactor, the ^{233}U chemically separated from the thorium, and fed back into another reactor as part of a closed fuel cycle.

In conventional PWR fuel assemblies could be arranged in such a way that a blanket of mainly thorium fuel rods surrounds a more enriched seed element containing ^{235}U, which supplies neutrons to the subcritical blanket. As ^{233}U is produced in the blanket it is also

burned there. This is the light water breeder reactor concept, which has been successfully demonstrated in the USA in the 1970s.

Currently, the thorium fuel cycle is again of interest to achieve improved proliferation resistance. The central seed region of each fuel assembly could have uranium enriched up to 20% in ^{235}U. The blanket could contain thorium with some ^{238}U, which means that any uranium chemically separated from it (to receive the ^{233}U) is not useable for weapons. Spent blanket fuel also contains ^{232}U, which decays rapidly and has very gamma-active daughter products. Plutonium produced in the seed element will have a high proportion of ^{238}Pu, generating a lot of decay heat and making it even more unsuitable for weapons than normal reactor grade Pu.

A variation of this concept is the use of fully homogeneous assemblies arranged in such a way that a set of them makes up a seed and blanket arrangement. If the seed fuel is metal uranium alloy instead of oxide, it has a better a heat conduction to cope with its higher temperatures. Seed fuel would remain three years in the reactor blanket for up to 14 years.

Potential concerns (disadvantages) of the Th-based NFC are:

- a general problem of any thorium-based fuel cycle is the high cost of fuel fabrication due partly to the high radioactivity of ^{233}U which is always contaminated with traces of ^{232}U;
- a similar problems exits in recycling thorium due to highly radioactive ^{228}Th;
- there is also some proliferation risk of ^{233}U;
- another issue are the technical problems (not yet satisfactorily solved) in reprocessing.

In conclusion, much development work is still required before the thorium fuel cycle can be commercialized, and the effort required seems unjustified while and where abundant uranium is available.

3.4.2.5. National research of a thorium NFC in existing reactors

In the following paragraphs several national programmes in regard to the use of a thorium NFC in operating reactors are laid out[12].

In Brazil, there is a long tradition of investigating the use of thorium, originally focusing on the application in heavy water reactors (HWRs). There was a programme started to further investigate the utilization of thorium in a PWR (Angra 1) [57].

In Canada, thorium fuel cycle studies for CANDU reactors have been performed by AECL since a long time. In the past, many techniques for thorium fuel fabrication have been investigated, hundreds of thorium-based fuel elements were fabricated (Fig. 3.10). A number of full-sized, thorium CANDU fuel bundles (U–Th, Pu–Th and pure Th) have been irradiated at full power for years in the NRU research reactor. Recently, a large code development programme was completed, allowing full-core fuel management.

[12] Long term aspects of the use of thorium NFC are discussed in Section 3.5.1.

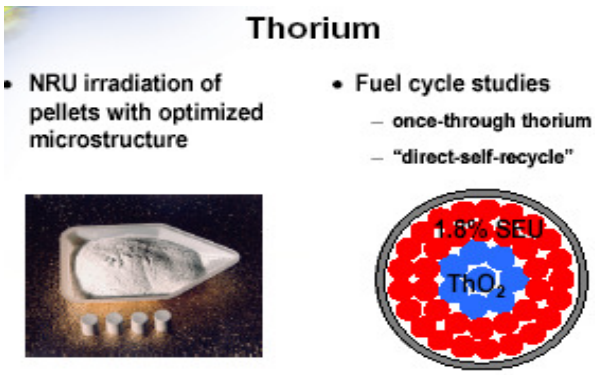

FIG. 3.10. Thorium fuel cycle studies in Canada.

The properties of spent thorium fuel as a waste product have also been studied in Canada. It was found that the high degree of chemical stability and the low solubility of thorium make irradiated thorium-based fuels attractive as waste forms for direct geological disposal. All together Th-based fuel for CANDU appears to be an attractive option for the future, while maintaining the advantages of the high refuelling flexibility of the CANDU core [57].

In Europe, some limited R&D efforts (partly EC funded), combined with in reactor irradiation (KWO, Germany), were underway to investigate the possibility to use thorium oxide together with UO_2 and MOX fuel in water cooled reactors. The major interest of R&D activities in Europe appears to focus on advanced technology solutions studied.

In India, Th-assemblies have been in use in several PHWR like the NPP Kakrapar-1 and 2, Kaiga-1 and 2, and Rajasthan-3 and 4, in order to flatten the neutron flux in the initial core during startup. India has made the commercial utilization of thorium for large scale energy production a major goal in its nuclear power programme (see Section 2.3.5). Th-fuel is also been used as blanket in the fast reactor design.

In Japan, there is no major activity on the Th fuel cycle for water cooled reactors, since the government intends to promote the U–Pu fuel cycle. Activities are restricted to basic studies mainly in universities.

In the Republic of Korea, so far there were no comprehensive plans or activities available yet regarding thorium fuel cycle applicable to water cooled power reactors.

Since the early 1990s, the Russian Federation has had a programme [57] to develop a thorium–uranium fuel, which more recently has changed to a particular emphasis on the utilization of weapons grade plutonium in a thorium–plutonium NFC. The studies confirmed that reactors of the WWER-1000 type can be used without essential changes to reduce weapon- and reactor grade plutonium surplus already in a first stage. The rate of plutonium utilization in WWER reactors would raise considerably compared to using MOX fuel. And there are no special problems expected to fabricate fuel based on mixtures of thorium and plutonium dioxides.

In the United Kingdom, BNFL appears unconvinced by current arguments for using thorium fuels in existing reactors except in special circumstances, and for operational reasons cannot use the existing uranium plant in the UK to fabricate them. However, through its connection with Westinghouse, the company was engaged in studies on thorium applications. In principle, it does not rule out more active future involvement subject to demonstration of clear benefits with technical and economic viability.

In the USA, the DOE is conducting four projects that involve the use of the thorium fuel cycle. All four projects are based on a once-through, proliferation resistant, high burnup, long

refuelling cycle use of thorium in a light water reactor. One of them is called Radkowsky Once-Through Thorium/Uranium Fuel Cycle (RTF, [62]) as one of the GEN IV NFC options. It investigates replacements for standard PWR assemblies consisting of two sub-regions within the pin cluster; an internal seed region and an outer blanket region. The seed region contains U–Zr (~20% enriched U) metallic alloy pins and the blanket region contains ThO_2UO_2 (~10%UO_2 at ~20% enrichment) pins. This NFC concept is considered by GEN IV as a near term option for use in existing PWR power plants (however, the results from the Generation IV studies indicate a potential starting of this NFC by 2020 and to fully adopt it by 2040).

3.4.3. The reduced moderation LWR-MOX recycle case

Reduced moderation (or high converter) LWRs are seen as a promising option for the next step of plutonium recycling after partial MOX loading in LWRs [63]. By taking advantage of LWR technology, they would contribute to early establishment of a multiple recycle system for plutonium, and thereby to a substantial reduction in the consumption of natural uranium.

Therefore, primarily in Japan (RMWR, Japanese water cooled FR in Section 2.5.5) but also in the Russian Federation (WWER-SCWR, see Section 2.5.7), concept and design studies were conducted to attain conversion ratios significantly above 1.0 and to effectively act as plutonium breeders for plutonium multiple recycling.

In the Japanese design study of the RMWR, various ideas such as a tight lattice fuel rod configuration, a short core design, and internal blankets for neutron absorption, have been introduced to increase as much as possible the conversion ratio from the fertile material of ^{238}U to the fissile material of ^{239}Pu, as well as to attain an automatic core power reduction characteristic even if the core power increases and generates bubbles, i.e. a negative void reactivity coefficient. As a result, some design concepts with various benefits have been obtained. One is a boiling water reactor (BWR) type concept with a conversion ratio of more than 1.0; this means that more fissile material than consumed can be produced in the core. The RMWR is self-sufficient in plutonium, because each recycle generation produces just sufficient plutonium to sustain the following generation. It follows that the mass of plutonium in each succeeding generation is constant and therefore, that the radiotoxicity of plutonium is essentially constant. The radiotoxicity from other elements, fission products and minor actinides, is similar to that of a conventional MOX cycle.

3.4.4. The advanced fuel cycle initiative (AFCI)

AFCI[13] is a broad R&D programme in the USA whose mission is to develop and demonstrate technologies that enable the transition from the current open NFC to an environmentally, socially, economically and politically acceptable advanced fuel cycle.

The primary AFCI goals are to develop fuel systems for Generation IV reactors and create enabling fuel cycle technologies, including fuel, cladding, fuel fabrication, separations, waste forms, and disposal technology, to reduce spent fuel volume, separate long lived, highly radiotoxic elements, and reclaim valuable energy from spent fuel.

The AFCI technologies will support both current and future nuclear energy systems. The AFCI is emphasizing the central role of systems analysis to define and assess the optimal deployment strategies, as well as the best possible transition from the current system to a future US nuclear fuel cycle.

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[13] http://www.afci.sandia.gov/

The transition from the current once-through fuel cycle to a closed fuel cycle is not yet fully defined, and is the subject of R&D under the AFCI programme. The stages in the development of and transition to advanced nuclear energy systems have been further elaborated and the definition of the important stages in the transition has been further refined.

The AFCI R&D programme focuses heavily of developing the technologies to allow for the fuel cycle transition. A systematic study of the fuel cycle transition options is being performed under the systems analysis task. This includes the development of transmutation criteria and transmutation analysis, analysis of repository impacts [64], and benefits of the fuel cycle options under scrutiny.

Fuels (see for example Ref. [65]) are being developed to allow the transition through the different phases (see Fig. 2.59 in Section 2.5.9) ranging from advanced fuels for LWR/ALWR plants, and fuels for advanced Generation IV reactors (both next generation nuclear plants and fast reactor systems for transmutation [66]). Actinides separation technologies continue to be the primary element of the AFCI, as advanced methods are required, tailored to each of the transition phases identified, and accounting for a variety of advanced fuels. Along with separations, development of storage forms for separated products is also a subject of current R&D.

The treatment of spent fuel from EBR-II (sodium-bonded) utilizing an electrometallurgical process remains a part of the AFCI programme and has continued at the facilities at ANL-W. Transmutation science and engineering is the remaining R&D area under the programme. It includes physics (nuclear data needed for transmutation systems), advanced materials for a transmutation system, and accelerator-driven transmutation systems (ADS). The materials development part overlaps also with the interests in the lead alloy fast reactor development under Generation IV. The ADS development is being done largely under international collaborations (such as TRADE[14], MEGAPIE[15], MARHA[16]).

3.4.5. Other near term NFC options

In the following sections some additional examples of near term NFC being studied are presented.

3.4.5.1. Two GEN IV once-through NFC

Within the Generation IV project (see Section 2.6.1) two once-through NFC are analyzed that combine LWR and HTR technology in order to increase the overall burnup:

- the mixed LWR-UO_2 and PBMR pebble bed once-through case;
- the mixed LWR-UO_2 and prismatic HTGR once-through case.

In the two NFC variants U oxide is used in both reactor types only. The beneficial outcome of both NFC is a moderately reduced waste, while the economic and the environmental situation (savings in raw material) is not significantly changed and also health effects to workers are small. But of course the advantage of these alternative NFC is that they would not fail to satisfy increasing power demands, due to the fact that they can use more or less proven technologies.

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[14] TRIGA reactor accelerator driven experiment, Italy, EU development project.

[15] Megawatt proton irradiation experiment, PSI Switzerland, SINQ accelerator, EU development project.

[16] Multipurpose hybrid research reactor for high tech application, ADS in MOL, Belgium, EU development project.

3.4.5.2. The LWR-MOX mono-recycle case

The LWR-MOX mono-recycle case with new reactors built for a 100% MOX core shows marginal benefits with regard to economy and environment criteria, but assumes large quantities of reprocessed MOX.

3.5. NFC technology with potential industrial deployment in 25–50 years

3.5.1. Advanced thorium fuel cycles

It appears that broad deployment of a commercial Th-NFC in countries other than India, depends on the progress of advanced reactor concepts.

3.5.1.1. Advanced reactor designs with thorium NFC

The following advanced nuclear power systems with a thorium NFC have the potential to be realized within the next 25–50 years:

- Advanced heavy water reactor (AHWR): The AHWR has been designed with Th-based fuel, using ^{233}U and Th as fuel matrix. If India is as successful in realizing its AHWR (Section 2.3.5) deployment programme in accordance to the planned schedule, others may be encouraged to follow, e.g. the advanced version of the CANDU reactor developed in Canada (Section 2.3.2).
- High conversion boiling water reactor (HCBWR): This concept is under consideration within the Generation IV advanced reactor concepts, using previous research on high conversion LWRs in the USA, Europe and Japan over the last several years [67], including advanced fuel designs such a metal matrix dispersion fuel.
- Advanced high temperature gas cooled reactor (HTGR): The French very high temperature (gas cooled) reactor VHTR (Section 2.4.3), which according to the French CEA is planned for 2020, shall operate with either TRISO UO_2/SiC fuel or (later) with ZrC coated particle fuel. This reactor type will be further developed in France to a gas cooled fast reactor (GFR), starting around 2025. Fast breeder reactors may use plutonium-based fuel to breed ^{233}U from thorium, and then advanced nuclear power systems will use the ^{233}U.

3.5.1.2. Research on long term application of thorium NFC

There is a potential for implementation of a thorium cycle for plutonium incineration in a molten salt reactor (MSR) system, called Thorium Molten Salt Nuclear Energy Synergetics (THORIMS-NES) [57]. This is investigated within a multinational study including experts from France, Japan and the Russian Federation. It is supposed to come up with a highly proliferation resistant nuclear fuel cycle. Based on an old ORNL study from 30 years ago, the idea is to combine thermal fission power stations and fissile producing accelerator driven molten salt breeders.

There are also studies performed in the Republic of Korea aiming at a Th fuel cycle concept for an accelerator driven system (ADS, Ref. [57]). This ADS programme is performed under the name HYPER (to be discussed further in Section 5.4.2.2) at Korea Atomic Energy Research Institute (KAERI) with the target to develop the elemental technologies for a subcritical system and build a small bench scale test facility (~5 MW(th)) after the year 2006. The fuel cycle concept for HYPER is supposed to utilize thorium in molten salt form to produce electricity as well as to transmute TRU elements. In this fuel cycle, fissile plutonium

isotopes in TRU will be incinerated to produce energy and to breed ^{233}U from thorium to be used later as fissile material.

As already mentioned before, there are some long term considerations [57] in the Russian Federation on the use of Th for Pu utilization. In these studies the use of Th is considered to be evolutional, passing through a series of stages. Firstly, reactors of the WWER-1000 type would be used. Spent thorium fuel assemblies would be stored to postpone decision on their reprocessing. In the next stages, fuel cycles are considered using both thermal and fast reactors that consume both U and Th. These studies include molten-salt thorium systems, critical and accelerator-driven, subcritical with continuous (or periodical) extraction for regeneration. Also a Th fuel cycle in a HTGR with fuel based on microspheric particle in a graphite matrix is investigated. For reprocessing of these fuels methods of dry fluoride technology could be applied. Last but not least, even the ^{233}U application for space reactors is part of these studies.

3.5.2. NFC for advanced fast reactors

There exist various fast reactor NFC concepts worldwide. Practical operating experience has been collected in the UK, France, Japan, the Russian Federation and the USA. Due to the long term experience with fast reactor technology in the Russian Federation, the Russian concepts should receive specific attention regarding realistic chances to end up with commercial maturity within the time frame of 25 to 50 years considered here. And within the various alternatives discussed in the Russian Federation the BREST concept [57] looks promising, at least under just technical aspects, and is therefore laid out in more detail in the following section.

3.5.2.1. Basic features of large scale nuclear power systems using the Russian BREST concept

Experts from the Russian Federation, Japan and the USA have investigated the possibility to design (U/Pu)N, lead cooled fast reactor cores with reactivity margin change less than 0.1% during the 10th fuel cycle without refuelling. The simple lead cooled cores loaded with mixed fuel of uniform composition and without blanket may be designed in such a way, that their operation requires no fuel shuffling until ultimate operating conditions for materials are achieved, while the reactivity variation versus burnup stays almost equal to zero.

The principal diagram of innovative fuel cycle based on a BREST reactor is shown in Figure 3.11.

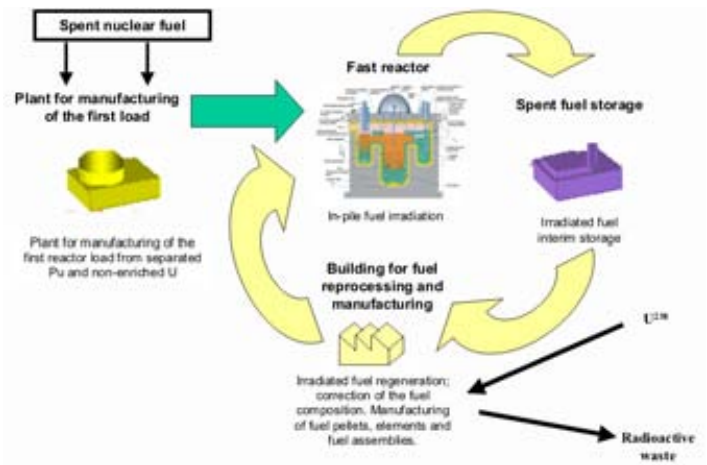


FIG. 3.11. Fuel cycle of BREST reactor with equilibrium U–Pu fuel.

Plutonium needed for the fabrication of the first BREST cores will be removed from temporary SNF storage places. Moreover, all SNF stored in pools at existing NPPs, will be reprocessed to recover plutonium.

The BREST-OD-300 on-site nuclear fuel cycle provides for a practically unlimited expansion of the available resource basis of nuclear power through organizing equilibrium composition nuclear U–Pu fuel circulation with the necessity of adding only an insignificant amount of depleted or natural uranium. Fuel elements of container type are used with fuel pellets made of mixed mono-nitride fuel placed inside the cladding with a gap filled with a Pb bond.

In this new fuel cycle it will be possible to establish a closed fuel cycle, at each stage of which plutonium will always circulate only in a mixture with ^{238}U, and chemical treatment of the irradiated fuel will not include the stage of plutonium separation from uranium. Thus this new technology will considerably strengthen the proliferation resistance of the fuel cycle.

It should be noted, that much lower amounts of minor actinides are accumulated in this closed fuel cycle, mitigating the radiation situation during fuel handling, and excluding the necessity of special actinide burners. Also much lower volume of radioactive waste is inherent to such a system.

For non-proliferation requirements it is necessary to keep uranium and plutonium inseparable in all reprocessing stages. The key objectives of fuel reprocessing are:

- Coarse fuel cleaning from FP, because fast reactor does not need more thorough fuel cleaning from FP.
- ^{238}U should be added to the fuel to compensate for the mass of extracted FPs.

In the fuel cycle under discussion, the ^{238}U added to the fuel during reprocessing, is burnt in the reactor. Pu and minor actinides are part of the fuel and circulate in the closed NFC as part of the highly active material.

In fact, fuel circulates within the reactor building and the adjacent fuel facilities. BREST fuel will never leave the closed system 'reactor + fuel building'. Cost analysis showed that the cost of the on-site fuel facilities would amount to 15% of the total cost of the plant with two BREST-1200 reactors.

The additional studies and tests of this technology will take some 10 to 15 years to develop and build an experimental reactor or a demonstration unit which can be put into demonstration operation in about 25 years.

3.5.2.2. Generation IV scenarios combining thermal with fast reactors

There are scenarios studied within the Generation IV initiative, which still stay in the range of just using U + Pu NFC technologies; however they combine thermal reactors with fast reactors:

- LWR U oxide once-through plants are combined with a fast burner reactor operating on a closed cycle.
- Fissile self-sufficient fast reactor closed fuel cycles are used in symbiosis with thermal once-through power plants.
- Already deployed fast reactors are converted from a burning (or fissile self-sufficient) operation to breeding at the appropriate time.

It should be noted that these scenarios assume starting of these systems not earlier than 2020–2025. There are more alternative fuel U/Pu NFC scenarios discussed in Generation IV studies, but with even longer time horizons.

CHAPTER 4.
FRONT END OF NFC

In the following sections history, current status and perspectives of the front end of NFC is presented covering uranium resources, mining & milling, conversion, enrichment and fuel fabrication[17].

4.1. Introduction

In most of the countries using nuclear energy some products and services within the NFC are purchased from several different sources, mainly with the purpose to profit from market competition. However, sometimes there are technical or economical reasons to buy a product or a service from a specific vendor.

Buying from several different vendors today is normally practiced in order to receive:

- uranium raw material, most in the form of U_3O_8;
- fuel assemblies or component of fuel assemblies.

Many countries are buying special services within their NFC because it is not economic for them to have a costly facility of their own for this service. This is normal practice in particular for enrichment services.

4.2. Uranium resources

The following subsections evaluate exploration, the availability and recovery costs, and unconventional resources of uranium, and existing inventories of fissile and fertile material.

4.2.1. Exploration of uranium

Successful worldwide exploration for uranium in the 1970s and early 1980s has played an important role in creating large and flexible production sources, and, as a consequence, decreasing uranium costs during these past two decades. This effect contributed significantly to the competitiveness of nuclear power generation in spite of increasing nuclear power plant erection cost.

According to the joint report (Red Book) by the OECD/NEA and the IAEA [9], the uranium exploration is geographically unevenly distributed and focuses on areas considered to be economically most attractive.

Worldwide 'non-domestic' exploration expenditures increased till 1994, decreased thereafter till 2003, and increased since 2003 sharply. Australia, Canada and France are the only countries spending considerable amounts of money into 'abroad (non-domestic)' exploration (in addition Switzerland is also exploring uranium but with a much smaller budget). Australia reported US $1.5 million in 2004, US $8.9 million in 2005, and US $4.6 million in 2006, and is expected to spend US $4.7 million in 2007. Canada reported US $2.5 million expenditures for non-domestic activities in 2002, US $9.6 million in 2004, US $53.9 million in 2005, US $124.6 million in 2006, and was expected to spend US $139 million in 2007, but focuses its activities on domestic activities. France reported exploration and development expenditures in Australia, Canada, Kazakhstan, Niger, Mongolia and the Russian Federation, amounting to a total of US $14.4 million in 2002, US $60 million in 2004, US $127.5 million in 2005, US $85 million in 2006, and was expected to spend US $115 million in 2007.

[17] Thorium was covered in the previous Section 3.4.2.

‘Domestic’ exploration expenditures strongly decreased between 1990 and 1994 followed by a considerable increase between 1994 and 1997, was going slightly up and down until 2003, and increased again considerably thereafter.

Canada continues to be the world leader in domestic exploration with annual expenditures in 2001 and 2002 of about US $16.2 and US $22.9 million, respectively, US $21.7 million in 2003, US $78.7 million in 2004, over US $432 million in 2006, and is expected to reach about US $458 million in 2005. A significant fraction of this amount is spent into advanced underground exploration, deposit appraisal activities and in care and maintenance preparation for projects close to production approval.

In the USA the total uranium exploration expenditures declined through 2001 to 2002 (US $4.8 million in 2001, US $0.35 million in 2002), but increased sharply in 2004 to US $59 million, US $77.8 million in 2005, and US $155 million in 2006.

In Central and South America, Argentina reported exploration expenditures of US $0.8 million in 2001, US $0.3 million in 2002, US $0.7 million in 2004, US $1.0 million in 2005, US $0.7 million in 2006, and is expected to reach US $0.65 million in 2007. Brazil did not report any exploration activities for 2001 and 2002, spent US $0.4 million in 2004, no expenditures in 2005 and 2006, and is expected to spend US $0.46 million USD in 2007. Chile reassessed regional geological information to improve knowledge of uranium potential spending US $0.1 million in 2004, 2005, 2006, and is expected to spend a similar amount in 2007.

In Western Europe and Scandinavia domestic exploration was only done in Finland reaching expenditures of US $0.21 million in 2004, US $0.8 million in 2005, US $1.8 million in 2006, and should reach US $3.5 million in 2007.

In central, eastern and south-eastern Europe, exploration activities are reported from the Russian Federation (mainly in sandstone amenable to in situ leaching and unconformity-related deposits; total expenditures are reported to amount to US $11.5 million in 2001, US $10.4 million in 2002, US $10.2 million in 2004, US $24. million in 2005, US $33.5 million in 2006, and are expected to reach US $63 million in 2007. Turkey explored granites and acidic intrusive rocks in 2003 and 2004 and will continue to explore in 2005 (spending about US $0.05 million). Ukraine continues to explore vein-type and unconformity-related deposits and spend US $1.7 million in 2001, US $1.9 million in 2002, US $3.4 million in 2003, US $4.3 million in 2004, US $4.8 million in 2005, US $6.2 million in 2006,and is expected to spend US $6.2 million in 2007.

In Africa, some exploration activities are reported from Egypt (about US $2.6 million in 2004, and US $1.7 million in 2005, 2006 and 2007), Niger (about US $6.4 million in 2004 to 2006), Namibia (about US $2 million in 2004 to 2006), and South Africa (about US $0.9 million in 2004, US $1.5 million in 2005, US $24.7 million in 2006, and about US $15 million in 2007).

In Asia there are active programmes underway in the following countries: India (US $14 million in 2004, US $16.5 million in 2005, US $16.4 million in 2006, and about US $23 million in 2007), the Islamic Republic of Iran (about US $3.7 million in 2004, reaching about US $8.8 million in 2007), Kazakhstan (US $11.8 million per year between 2000 and 2002, US $4.3 million in 2003, US $0.7 million in 2004, US $1.2 million in 2005, US $8.5 million in 2006 and expected US $26.3 million in 2007), and Uzbekistan (about US $17 million in 2004, and US $22.1 million per year from 2005 to 2007).

In Indonesia, the Philippines and Vietnam exploration activities are maintained at low level.

In East Asia, China reported exploration expenditures of US $6.0 million and US $7.3 million in 2001 and 2003 respectively, US $9.5 million in 2004, US $13.5 million in 2005, and US $25.5 million in 2006. In 2007, exploration expenditures are expected to amount to US $33.6 million. Exploration appears to continue in Mongolia, but no details are reported.

In the Pacific Region exploration continues in several areas of Australia on low to medium level in terms of expenditures, i.e. US $2.5 million in 2001, US $3.0 million and 2002, US $10 million in 2004, US $31 million in 2005, US $61 million in 2006, and expected US $70 million in 2007.

In Figure 4.1 the development of worldwide domestic and abroad exploration expenditures between 1994 and 2007 is depicted [9].

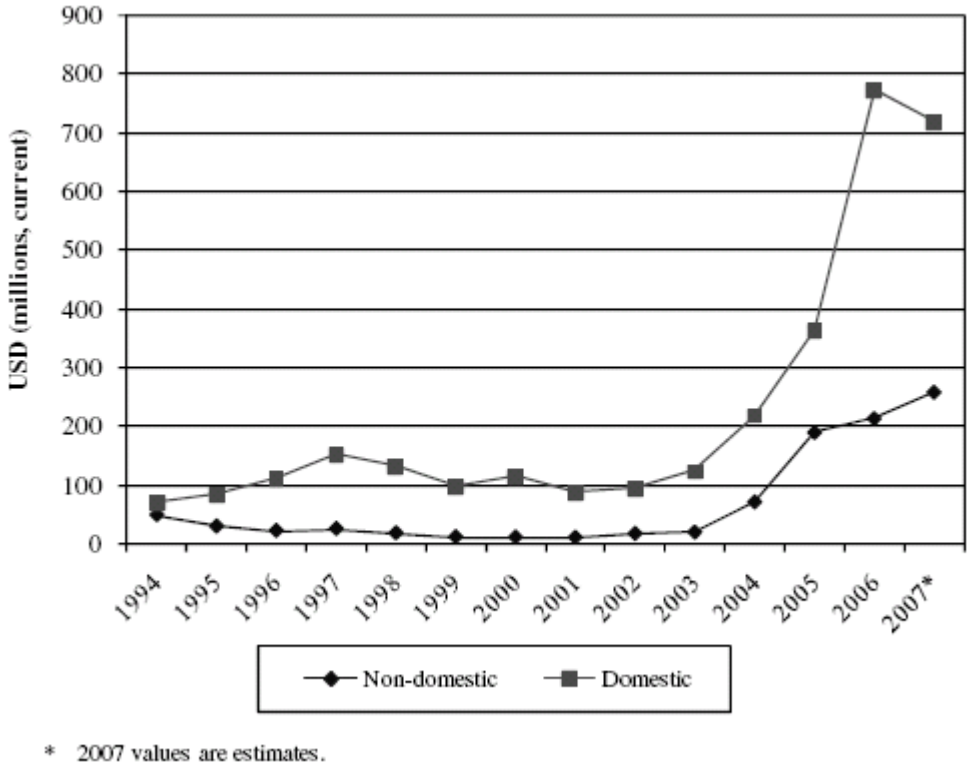


FIG. 4.1. Exploration expenditures worldwide (Ref. [9] published 2008).

After a continuous decrease between 1997 and 2003, worldwide exploration has increased significantly thereafter. Thus, Figure 4.1 clearly demonstrates that mining industry is reacting to the projected increase of demand and price of uranium (to be discussed further in Section 4.3) and consequently increasing exploration.

4.2.2. Availability and recovery costs of uranium

Following the joint report (Red Book) by the OECD/NEA and the IAEA [9], identified uranium resources (formerly as part of the conventional[18]*l* resources KCR) consist of:

- reasonably assured resources (RAR);
- inferred resources (formerly called EAR-I).

[18] Non-conventional resources are discussed in Section 4.2.3.

Table 4.1 shows that the identified resources primarily in the range of <US $130/kg significantly increased between 2001 and 2007. The increase between 2001 and 2003 is mainly the result of increased resources reported by Australia, Canada, Kazakhstan, Niger and the Russian Federation, as well as for the first time by China and India. Between 2003 and 2005 the increase is due to reported changes of resources in Australia, Brazil, Kazakhstan, Niger, South Africa and Ukraine.

The overall increase of identified resources between 2001 and 2003 of about 655000 t·U equals to about 10 years of 2002 global uranium requirements (for the existing fleet of reactors).

TABLE 4.1. URANIUM RESOURCES 2001–2007 IN 1000 t·U (REF. [9] PUBLISHED 2008)

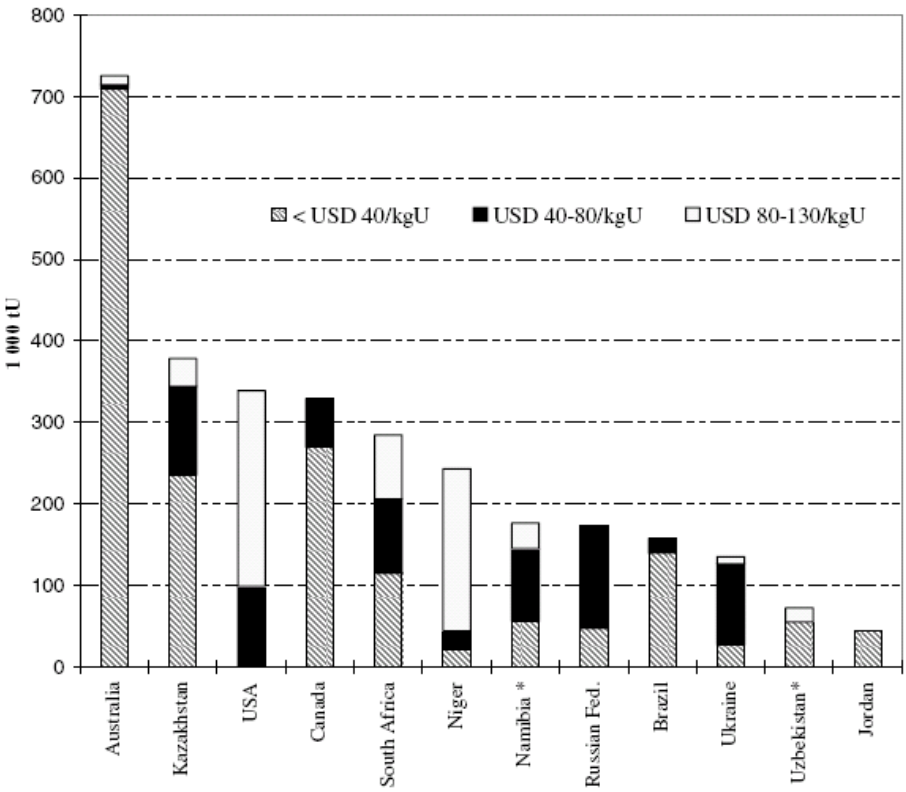
Resource category	2001	2003	2005	2007
Identified resources *				
<US $130/kgU	3933	4548	4743	5469
<US $80/kgU	3107	3537	3804	4456
<US $40/kgU**	>2086	>2523	>2746	2970
RAR				
<US $130/kgU	2853	3169	3297	3338
<US $80/kgU	2242	2458	2643	2598
<US $40/kgU**	>1534	>1730	>1947	1766
Inferred resources				
<US $130/kgU	1080	1419	1446	2131
<US $80/kgU	865	1079	1161	1858
<US $40/kgU**	>552	>793	>799	1204

* Identified resources are the sum of RAR and inferred resources.

** Recourses in the cost categories of <US $40/kgU are higher than reported, however, several countries have indicted that either detailed estimates are not available, or the data are confidential.

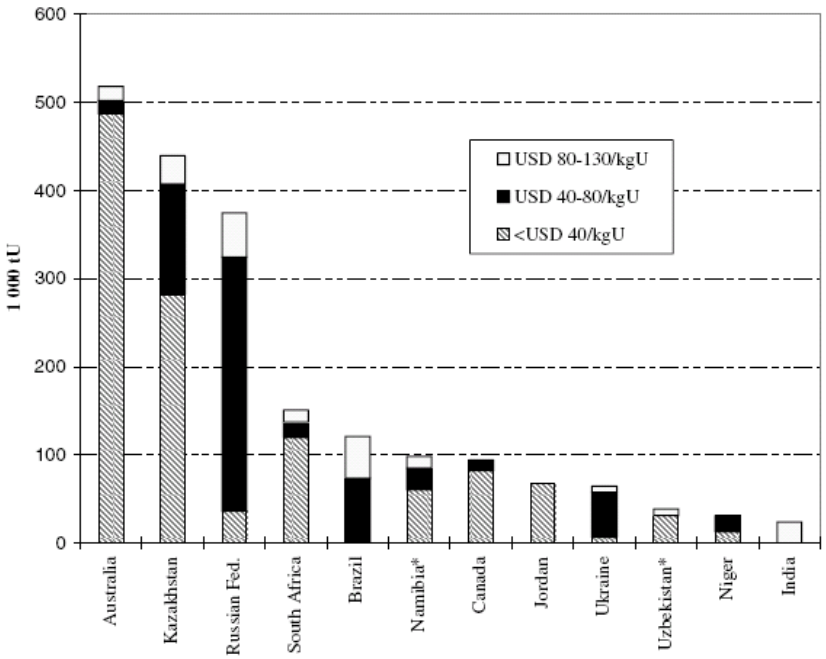
Current country-by-country estimates could be looked up in the most recent version of the mentioned Red Book (currently, Ref. [9]). Nevertheless, in Figures 4.2 and 4.3 for countries with major resources an overview is given on the distribution of RAR and inferred resources, respectively.

In addition to the identified uranium resources (presented above) there are two more categories defined in the Red Book [9] called prognosticated and speculative resources with a reported (2007) value in the cost range of <US $130/kg of 2.8 and 4.8 million t·U, respectively.



* Secretariat estimate.

FIG. 4.2. Distribution of reasonably assured resources (RAR) among countries with major resources (Ref. [9).



* Secretariat estimate.

FIG. 4.3. Distribution of inferred resources among countries with major resources [9].

4.2.3. Non-conventional uranium resources

The non-conventional resources consist of uranium in sea water and in phosphates. The total amount of uranium dissolved in seawater at a uniform concentration of 3 mg U/m^3 in the world's oceans is estimated to be 4.0 billion tonnes. In Reference [5] costs of uranium from seawater are estimated to be on the order of US $300/kg U. In phosphates 22 million tonnes are reported with recovery costs between US $60 and US $100/kg U.

4.2.4. Inventories of recyclable fissile and fertile material

A recent study from NEA [68] has dealt with the additional available resources of fissile and fertile material in stock worldwide at the end of 2005. The following Table 4.2 shows such materials from the different sources.

TABLE 4.2. INVENTORY OF SEPARATED RECYCLABLE MATERIALS [68]

Source	Quantity (tHM)	Nat.U equ. (10^3 tU)	Reactor-years of supply*
Ex-military HEU	230	70	420
Ex-military Pu	70	15	90
Pu	320	60	380
RepU	45 000	50	300
Enrichment tails	1 600 000	450	2 650

** Based on a 1000 MWe LWR operating at 80% load factor.*

In case the entire inventory in the table above would be recycled this would provide the existing reactor fleet with an additional 10 years of fuel supply. If, further, all accumulated spent fuel would be reprocessed and recycled, some 1700 tonnes of Pu and 190 × 10^3 tonnes of natural uranium equivalent would become available, corresponding to about 7 years of supply for the current fleet of nuclear power plants in operation.

Thus, these inventories of recyclable fissile and fertile material represent a large potential energy resource.

4.3. Mining and milling of uranium

In the following subsections, the different processes of uranium mining, suppliers and customers, and worldwide production and demand currently and projected are laid out.

4.3.1. Types of mining

Commercial uranium mining is considered to be attractive if the concentration of uranium in the ore is above ~0.6% U. Uranium is extracted from the ground by three basic processes:

- Underground mining, this is a traditional process with shafts sunk into the earth in order to gain access to the uranium ore.
- Open pit mining, this of course needs the ore being close to the surface.

In both cases above, the ore is transported to a mill in which the uranium is separated from the ore. The third process is:

- In situ leaching (ISL), there the uranium is extracted from the ore in situ by the use of a leaching solution.

Uranium is also recovered as a by-product from the mining of other minerals, such as gold, copper and phosphate.

Once the uranium ore has been extracted in underground or open pit mining, it is processed in a mill where the uranium is leached from the ore using either an acid or an alkaline leaching solution. The uranium is recovered from this solution, or from ISL solutions, using an ion exchange or solvent extraction process. The common mill product is a uranium oxide concentrate U_3O_8, named yellow cake. The yellow cake is usually heated to drive off impurities, thus increasing the U_3O_8 concentration.

As shown in Table 4.3, open-pit and underground mining (and conventional milling) although declining continue to be the dominant uranium production technologies, accounting for about 70% of total production in 2002, 67% in 2005, 65% in 2006, and expected 61% in 2007.

TABLE 4.3. PERCENTAGE OF DISTRIBUTION OF WORLD URANIUM MINING BY METHOD [9]

Mining method	2001 (%)	2002 (%)	2003 (%)	2004 (%)	2005 (%)	2006 (%)	2007*** (%)
Open pit mining	26.1	26.8	29.8	27.6	28.1	24.2	23.7
Underground mining	44.2	43.1	41.6	39.3	37	40	37.7
In situ leaching	15.5	18.3	18.4	19.8	20	25	27.7
Heap leaching	1.2	1.7	1.9	2.0	2.2	2.2	2.4
In place leaching *	0.1	0.1	0.0	0.0	0.1	0.1	0.1
Co-product/by-product	12.4	9.1	9.7	11.1	10	9	8.4
Other methods **	0.5	0.8	0.5	0.2	0.2	0.2	0.1

* Also known as stop leaching or block leaching.
** Includes mine water treatment and environmental restoration.
*** Expected.

Uranium recovery from seawater

In Japan, uranium recovery from seawater has been studied as a supply option for a very long time. JAEA has been developing adsorption technologies using amidoxine adsorbents for uranium recovery from seawater. A pilot scale experiment has been conducted by JAEA using a uranium-specific non-woven fabric packed in an adsorption cage 16 m^2 in cross-sectional area and 16 cm in height. Three adsorption cages were submerged in the Pacific Ocean at a depth of 20 m at 7 km offshore of Japan. The three adsorption cages consisted of stacks of 52000 sheets of the uranium-specific non-woven fabric with a total mass of 350 kg. The total amount of uranium recovered by the non-woven fabric was >1 kg in terms of yellow cake during a total submersion time of 240 days in the ocean. An adsorption method using polymeric adsorbents capable of specifically recovering uranium from seawater was evaluated to be economically feasible.

Uranium recovery from phosphates

The technology of recovering uranium from phosphates is mature, but due to the high costs not commercially applied up till today.

4.3.2. Suppliers and customers of uranium

In Table 4.4, a general survey is given for countries where they are getting the U_3O_8 concentrate and/or to where they are providing it to.

TABLE 4.4. URANIUM (U_3O_8): SUPPLIERS AND CUSTOMERS [49]

Customer	Supplier					
	Australia	Canada	USA	FSU	Domestic	Others
Argentina					x	x
Armenia				x		
Belgium	x	x	x			x
Brazil					x	
Bulgaria				x		
Canada					x	
China					x	
Czech Republic				x	x	
Finland	x	x	x	x		x
France	x	x	x	x	x	x
Germany	x	x	x	x		x
Hungary				x		
India					x	x
Japan	x	x	x			x
Kazakhstan					x	
Republic of Korea	x	x	x	x		x
Lithuania				x		
Mexico		x				x
Netherlands						x
Romania					x	
Russian Federation					x	
Slovakia				x		x
Slovenia		x	x	x		
South Africa			x		x	
Spain	x	x		x	x	x
Sweden	x	x	x	x		x
Switzerland	x	x	x	x		x
Ukraine				x	x	
UK	x	x	x	x		x
USA	x	x		x	x	x

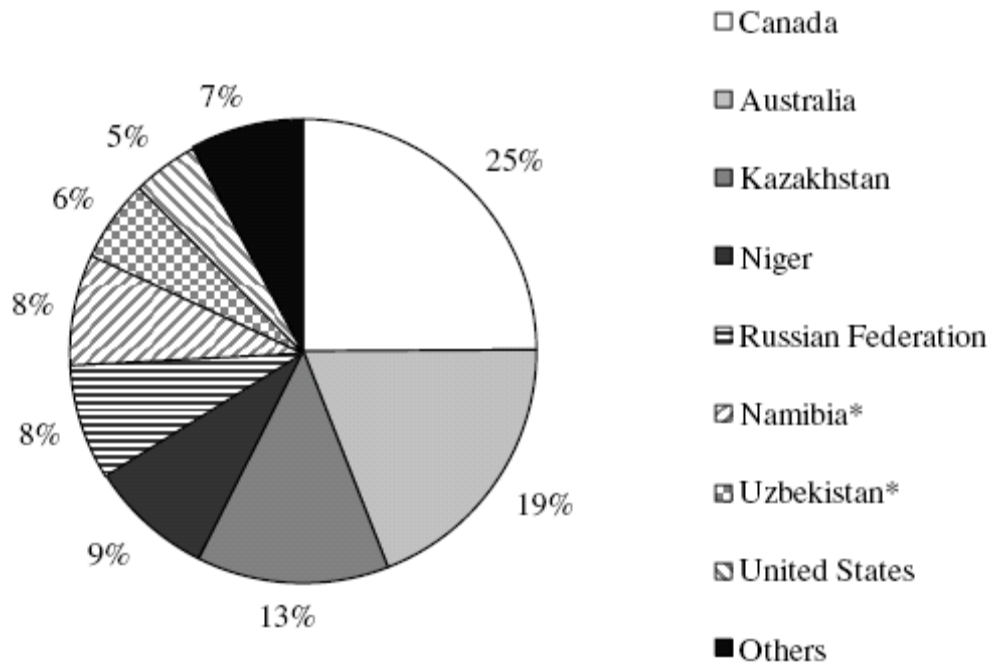
The following information in Sections 4.3.3 to 4.3.6 is taken from the Red Book (Ref. [9] published 2008).

4.3.3. Current worldwide uranium production

At present, the main uranium ore producing countries (see Fig. 4.4) are Australia (19% of world production in 2006 of 39603 t·U), Canada (25%), Kazakhstan (13%), Namibia (8%), Niger (9%), the Russian Federation (8%), the USA (5%), and Uzbekistan (6%). 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%), and Ukraine (2.0%).

World uranium production increased by 11% from 36050 t·U in 2002 to 41943 t·U in 2005, decreased slightly in 2006 to 39603 t·U and is expected to increase further in 2007 to >43000 t·U.

Within OECD countries, production increased slightly from 20114 t·U in 2002 to 21956 t·U in 2004, but decreased thereafter to 19705 t·U in 2006.



* Secretariat estimate.

FIG. 4.4. Uranium production in 2006 (39603 t·U).

4.3.4. Worldwide uranium demand

World annual uranium demand was estimated at about 66815 t·U in 2002, 68435 t·U in 2003, 67320 t·U in 2004, 66500 t·U in 2006, and estimated to increase to 69110 t·U in 2007. The regional distribution of demand in 2006 is shown in Figure 4.5.

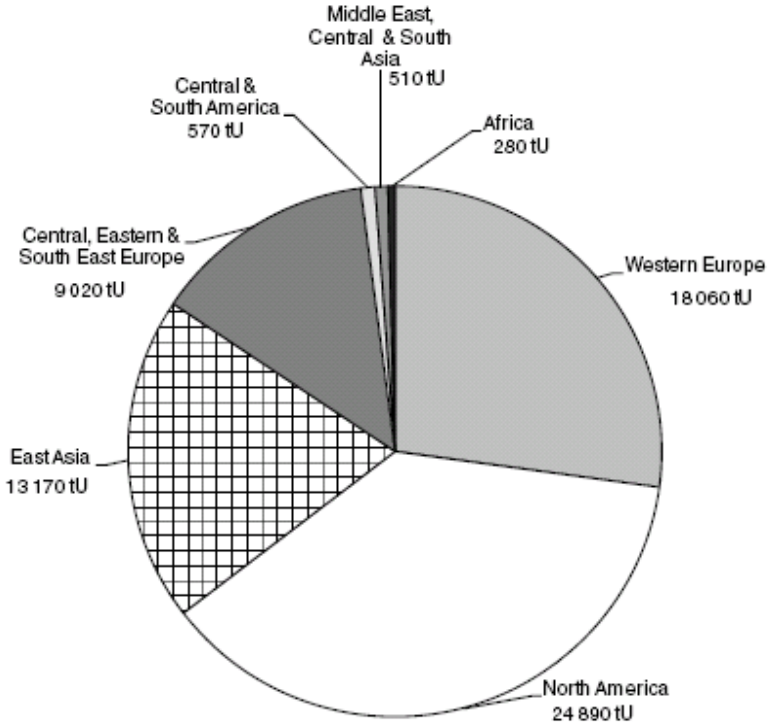


FIG. 4.5. 2006 World uranium requirements per region [9].

Details on the regional and country-by-country requirements could be looked up in the NEA/IAEA Red Book [9].

4.3.5. Current relationship between worldwide uranium demand and supply

Following the political and economic reorganization in Eastern Europe and the former Soviet Union in the early-1990s, major steps toward the development of an integrated commercial world uranium market have been taken. As a consequence there has been greater availability of uranium supplies from Kazakhstan, the Russian Federation and Uzbekistan.

Despite the increased availability of information regarding the amount of uranium held in inventory by utilities, producers and governments, uncertainty remains regarding the magnitude of these inventories (especially in the Russian Federation) and the availability of uranium from other sources. This uncertainty continues to have a significant influence on the uranium market.

Primary (from mines) uranium production alone is insufficient to meet world uranium requirements of nuclear power. In 2004, world uranium production (36042 t·U) provided only about 54% of the world reactor requirements (67320 t·U), and in 2006 world production reached about 40000 t·U covering about 62% of world reactor requirements (66500 t·U).

In OECD countries, the 2004 production of 21956 t·U provided only about 39% of the demand of 55610 t·U (Fig. 4.6), decreasing further to about 35% in 2006.

However, remaining requirements of uranium are met by secondary sources including stockpiles of natural and enriched uranium (both civilian and military in origin), the reprocessing of spent fuel and the re-enrichment of depleted uranium tails, i.e. the uranium supply and demand remained in balance.

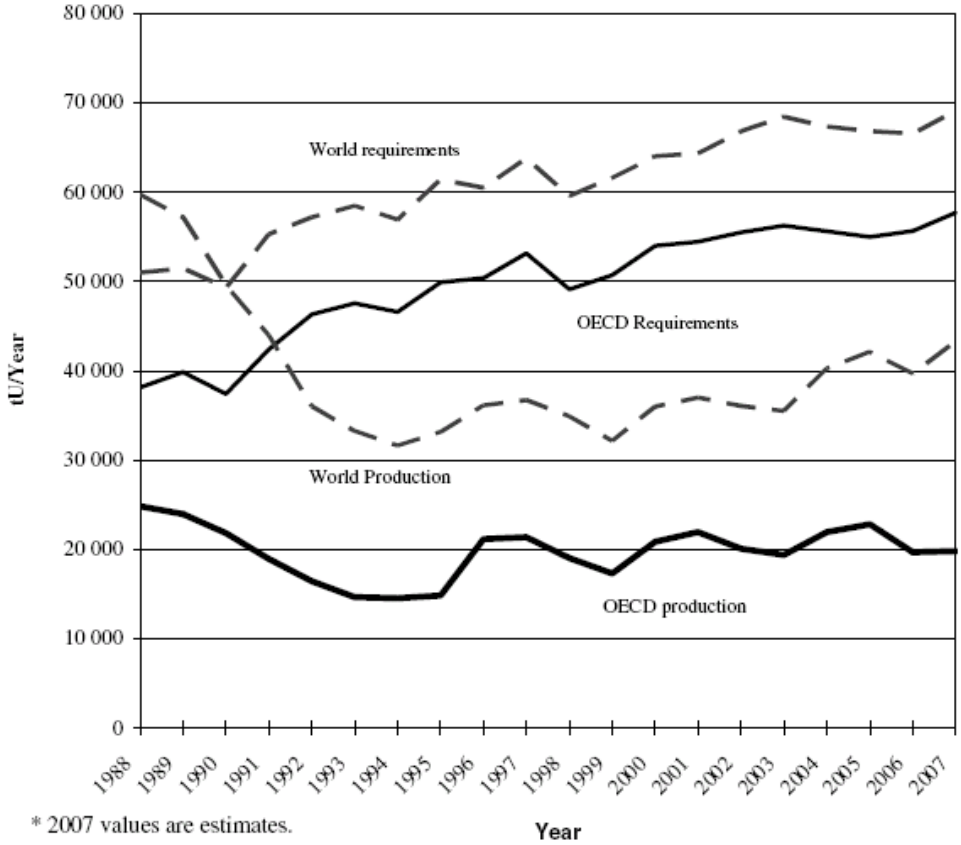


FIG. 4.6. OECD and world uranium production and requirements [9].

4.3.6. Projection of worldwide uranium demand and supply till 2030

Forecasts [9] of installed nuclear capacity (and corresponding uranium demand) are uncertain but generally point to future growth. Installed capacity is projected to grow from 369 GW(e) net at the beginning of 2005 to about 449 GW(e) net (low case) or 533 GW(e) net (high case) by the year 2030. Nuclear capacity projections also vary considerably from region to region, e.g. East Asia is supposed to experience the largest increase and Western Europe might see a significant decrease due to existing nuclear energy phase out plans.

Market conditions are the primary driver of decisions to develop new or expand existing primary uranium production centres. After a long period with falling market prices of uranium, these prices have increased since 2003 and are supposed to increase even further. Therefore, the supply forecast has changed dramatically from 2003 to 2005, i.e. significant new uranium production has been planned or committed (Fig. 4.7).

Nonetheless, a critical examination of the projected uranium production capability through 2030 indicates that secondary sources will continue to be needed to meet projected demand for a couple of years. In 2030, planned capability from all reported Existing and Committed Production centres based on resources recoverable at a cost of <US $80/kgU — although between 2010 and 2017 the projected production capability exceeds demand —- are adequate to cover about 89% of the low case uranium requirements but only about 68% of the high case. Adding in Planned and Prospective Production centres would satisfy the low case till 2030 and the high case demand through 2028. Moreover, it is to be noted that world production has never exceeded 89% of production capability. Hence, additional primary production and/or additional secondary supply would be needed.

Additionally, after 2013, when secondary sources of uranium are expected to decline in availability, reactor demand of uranium will have to be met increasingly by primary production. Therefore, primary production capability will need to be increased by expanding existing production centres, opening new production centres or through a combination of the two.

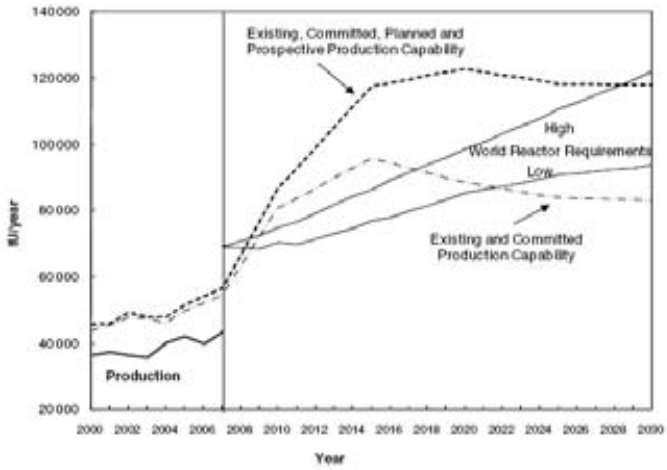


FIG. 4.7. Projected annual world uranium production capability through 2030 compared with projected world reactor requirements[19] [9].

[19] Includes all existing, committed, planned and prospective production centres supported by RAR and Inferred Resources (formerly called EAR-I) recoverable at a cost of <US $80/kgU.

4.3.7. Environmental aspects of uranium mining and milling

In all countries, there is increasing awareness of the need for environmental protection [70] during mining and milling of uranium (and thorium) ore. Large programmes have been underway in several countries to clean up wastes from closed U mines and mills. For new projects, companies must prepare and submit an environmental impact assessment to regulatory authorities. This normally consists of detailed planning for the life of the project, including relevant safety and environmental aspects.

The environmental assessment process may involve public hearings that provide for discussion of socioeconomic impacts and the concerns of stakeholders in the communities affected by the project. This holds in particular for mining and milling in Canada and in Australia.

One specific concern relates to the long term management of wastes resulting from mining and milling. In 2000, an International Symposium on The Uranium Production Cycle and the Environment was held by the IAEA in cooperation with all internationally institutions relevant in this area [71]. Based on the results of this symposium, IAEA developed and released in 2002 a Safety Guide on Management of Radioactive Waste from the Mining and Milling of Ores [72].

4.4. Conversion of uranium

4.4.1. Types of conversion

The word 'conversion' is used in NFC in two different meanings. In this stage, it refers to the processes of purifying the uranium concentrate U_3O_8 and then converting it to the chemical form needed in the next stage of the fuel cycle.

Basically, conversion to three such chemical forms is normally in use:

- conversion to U-metal;
- conversion to uranium oxide (UO_2);
- conversion to uranium hexafluoride (UF_6).

UF_6 is the product normally used at this stage in today commercially applied uranium NFC. This holds true in particular when the next step of the NFC is the enrichment of ^{235}U, because UF_6 is easily converted to a gas at moderate temperatures. Thus, UF_6 is the intermediate product for all fresh fuel used in LWRs, AGRs, etc. whenever enriched fuel is used.

For the conversion from U_3O_8 to UF_6 the uranium concentrate is dissolved in nitric acid, then filtered, and treated with chemical solvents. The resulting uranyl nitrate (>99.95% pure) is then reconverted to uranium oxide and this, in turn, is converted to readily evaporable UF_6, to be used in enrichment facilities.

Re-conversion of depleted uranium hexafluoride to U_3O_8 currently is practiced in France and planned in the United States of America in order to take care of the growing stock of depleted uranium.

Five countries provide this conversion of U_3O_8 to UF_6 at commercial scale: Canada, France, the Russian Federation, the United Kingdom and the USA.

In the CANDU fuel cycle, up to today natural uranium oxide is used as fuel. For this purpose uranium is purified and converted directly to UO_2 or UO_3: in Argentina, Canada, China, India and Romania.

The Magnox fuel cycle uses natural uranium metal as fuel. Since this fuel is only used in British reactors, only in the UK a commercial facility for the production of U metal fuel is operating.

4.4.2. Suppliers and customers of conversion

In Table 4.6 a general survey is given on countries where they are buying and to whom they are providing conversion services.

TABLE 4.6. CONVERSION TO UF_6: SUPPLIERS AND CUSTOMERS [49]

Customer	Supplier					
	Canada	France	UK	USA	Russian Federation	Domestic
Armenia					x	
Belgium	x	x	x	x		
Brazil			x	x		
Bulgaria					x	
China						x
Czech Republic	x	x			x	
Finland	x	x			x	
France				x	x	x
Germany	x	x	x	x	x	
Hungary					x	
Japan	x	x	x	x		
Republic of Korea	x	x	x	x	x	
Lithuania					x	
Mexico	x	x				
Netherlands			x			
Russian Federation						x
Slovakia					x	
Slovenia		x		x	x	
South Africa		x				
Spain	x	x	x	x	x	
Sweden	x	x	x	x	x	
Switzerland	x	x	x	x	x	
Ukraine					x	
UK	x					x
USA	x	x	x		x	x

Additionally, Argentina, India and Romania are producing UF_6 for domestic use.

In Table 4.7 commercial conversion plants are listed together with their typical processes.

TABLE 4.7. COMMERCIAL CONVERSION PLANTS AND PROCESSES

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

Source: Roadmap Fuel Cycle Assessment Report, Generation IV International Forum December 2002.

4.4.3. Environmental and safety aspects of conversion

Conversion plants are special chemical facilities; they handle very aggressive chemicals (e.g. F, HF), but they do not produce significant amounts of radioactive effluents. Therefore, the environmental and safety criteria are comparable to those applied in the corresponding chemical industry handling similar aggressive products.

4.4.4. Future perspectives of conversion

As long as enriched ^{235}U is the backbone of the existing and the future fuel cycles, there will be no fundamental changes of technology necessary, besides normal technical development in the chemical industry for increasing the effectiveness of the processes and adjusting the technology to future environmental and safety criteria.

4.5. Enrichment of uranium

4.5.1. Types of enrichment

Today two main technologies are practiced for the commercial enrichment of uranium:

- Gaseous diffusion (GD): separation is achieved by using the faster diffusion rate of ^{235}U through a porous membrane as compared to ^{238}U. The uranium is used in the form of

gaseous UF_6. This process is very energy intensive and requires very large plants for commercial operation.

- Centrifuge enrichment, separates the lighter ^{235}U by applying very high rotational speeds, again using the form of gaseous UF_6. One of the economic advantages of this technology is that the facility can be developed in a modular way, allowing expansion of the facility according to demand.

Since reprocessed uranium (REPU) contains isotopes, which are difficult to handle at a diffusion plant, centrifuge technology is used for enrichment of REPU, also due to the low inventory and modular design of centrifuges.

A LWR with 900 MW(e) output needs about 20 t of heavy metal (HM) fuel annually. In case it is enriched to about 4% of ^{235}U it requires roughly 160 t of natural uranium UF_6 feed and the expenditure of about 100000 SWU[20].

4.5.2. Uranium enrichment suppliers and customers

Figure 4.8 shows the world supply of enrichment services (USEC = USA, Urenco = Netherlands/Germany/UK, AREVA = France, Rosatom = Russian Federation, others = Argentina, Brazil, China, India, Japan, Pakistan).

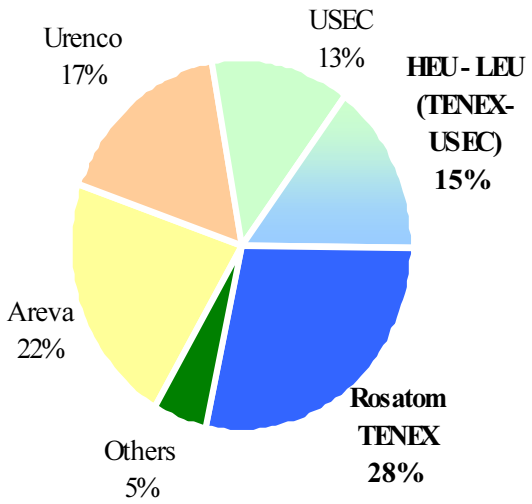


FIG. 4.8. Coverage of the world SWU demand by leading suppliers.

[20] SWU means Separative Work Unit, a measure of the amount of work performed in separating the two isotopes ^{238}U and ^{235}U.

In Table 4.8 the commercial enrichment plants are listed together with their applied processes.

TABLE 4.8. COMMERCIAL ENRICHMENT FACILITIES

Country	Place	Operator	Capacity 10^3 SWU/a	Enrichment process	First operation
Argentina	Pilcaniyen	CNEA	20	Diffusion (SIGMA)	2007
Brazil	Resende	INB	115	Centrifuge	2006
China	Shaanxi/Lanzhou	CNNC	1000	Centrifuge	2005
France	Tricastin (George Besse)	Eurodif	10800	Gaseous diffusion	1979
Germany	Gronau	Urenco	1700	Centrifuge	1985
Japan	Rokkasho-Mura-1	JNFL	600	Centrifuge	1992
	Rokkasho-Mura-2	JNFL	450	Centrifuge	1997
Netherlands	Almelo	Urenco	2500	Centrifuge	1973
Russian Federation	Angarsk	Minatom*	1400	Centrifuge	1954
	Ekaterinburg	Minatom	10000	Centrifuge	1949
	Krasnoyarsk	Minatom	2900	Centrifuge	1964
	Seversk	Minatom	5700	Centrifuge (recycled U)	1950
UK	Capenhurst	Urenco	3100	Centrifuge	1976
USA	Paducah	DOE/USEC	11300	Gaseous diffusion	1954

*Name of Minatom changed to Rosatom.

In Table 4.9 a general survey is given on countries where they are buying and to whom they are providing enrichment services.

TABLE 4.9. URANIUM ENRICHMENT: SUPPLIERS AND CUSTOMERS [49]

Customer	Supplier				
	Eurodif	USA	Urenco	Russian Federation	Domestic
Argentina				x	x
Armenia				x	
Belgium	x	x	x	x	
Brazil		x	x		
Bulgaria				x	
China					x
Czech Republic		x		x	
Finland	x		x	x	
France		x	x	x	x
Germany	x	x	x	x	
Hungary				x	
India					x
Japan	x	x	x		x
Kazakhstan				x	
Republic of Korea	x	x	x	x	
Lithuania				x	
Mexico		x			
Netherlands		x	x		
Russian Federation					x
Slovakia				x	
Slovenia		x		x	
South Africa	x	x	x	x	
Spain	x	x		x	
Sweden	x	x	x	x	
Switzerland	x	x	x	x	
Ukraine				x	
UK		x	x	x	
USA	x		x	x	x

In the following section, as an example, the enrichment services of the Russian Federation are described in some more detail.

4.5.2.1. Uranium enrichment in the Russian Federation

The Russian Federation possesses a powerful isotope separation industry (~40% of the world market) based on the currently most cost-efficient centrifugal technology of isotope separation (Refs [73–75]).

The total centrifuge capacity of four separation plants in the Russian Federation is estimated at ~20 mln SWU with a load of ~95%. This capacity includes the enrichment plants Ural Electrochemical Integrated Works (UEIW), Electrochemical Plant (ECP), Siberian Chemical Integrated Works (SCIW) and Angarsk Electrolysis Chemical Integrated Works (AECIW).

The enrichment capacity in the Russian Federation surpasses its nuclear power needs. According to 2000 data, 40.8% of the enrichment capacity was used to enrich uranium for the domestic nuclear power needs; 12.9% was used to reprocess dumps of foreign customers;

28.9% was used to reprocess high and low enrichment uranium and 17.4% was used for export orders.

The Russian Federation stopped to produce weapons grade uranium since 1989. Isotope dilution of weapons grade uranium is based on using standard equipment and technologies of uranium isotope separation plants. By 2004, 170 t out of 500 t of highly enriched uranium have been reprocessed in the framework of the Agreement on Using Highly Enriched Uranium.

The major field of development in the uranium enrichment area is the improvement of gas centrifuges. They are highly reliable: the specified service life for the latest Generation 6 centrifuges is 25–30 years and the failure rate of Generation 6 machines is less than 0.1% a year. Six generations of gas centrifuges have been developed at OKB LKZ (OKBM, currently Tsentrotekh–ECP, St. Petersburg). The first pilot centrifuge plant was started up in 1957, and it was in 1997 that the first unit equipped with the UEIW-designed Generation 7 centrifuges was put into operation.

Presently, a programme of major activities to upgrade separation facilities for a period of up to 2010 is being implemented in the Russian Federation. As part of this programme, it is planned to replace all old centrifugal uranium separation machines with the latest Generation 7 machines, and subsequently launch serial production and convert to operation of a subcritical PGTs-8 machine. A supercritical NPGTs-9 model, a fundamentally new machine for Generation 9, should be developed by 2008–2010. It is planned to raise the total separation capacity by 34% by 2010.

Apart from the development of a centrifugal enrichment technology, laser enrichment methods are investigated, primarily at the Russian Research Centre of Kurchatov Institute. The scope of studying laser enrichment methods in the Russian Federation was much smaller as compared with the USA. However, the large scale implementation of laser uranium enrichment in the Russian Federation is unlikely in near future.

4.5.3. Environmental, safety and non-proliferation aspects

Depleted uranium (DU) is a by-product of enrichment. For a typical LWR, between 6 and 8 tonnes of DU are produced per tonne of fresh fuel. Management strategies for this material are different from country to country. France is converting the DU to a stable oxide for long term storage and possible eventual reuse as fuel in breeder reactors, or for re-enrichment for extraction of additional fissile ^{235}U. The US Nuclear Regulatory Commission has stated that some type of deep disposal would be required if it were to be disposed of. However, no decisions have been made so far.

Similar to the conversion technology the safety criteria of corresponding chemical industry risks are adequate. Of course, there has to be an adequate safeguard control on the enriched materials.

4.5.4. Future perspectives of enrichment technology

Technically alternative separation processes have been and still are under consideration. In particular the old diffusion separation plants need to be replaced soon. Laser isotope separation could be a candidate for the replacement. However, development was strongly reduced in some countries, like France, and it was stopped completely in the USA.

Commercially, due to large overcapacities (15–20 million SWU) worldwide, the market for enrichment services is expected to show little growth over the next two decades. Driven

primarily by increased demand in Asia, world demand may increase to about 37 million SWU by 2005, but afterwards will remain constant until 2010.

The USA and France are going to replace their facilities with diffusion technology by centrifuges in the near future.

The Russian Federation has offered recently to establish a multilateral enrichment centre under the auspices of the IAEA (see Annex B).

4.6. Fuel design, fabrication and operation

4.6.1. LWR fuel for BWR, PWR and WWER

4.6.1.1. Short historical review

LWR fuel technology exists since more than 40 years. To achieve the present high level of reliability and economics a remarkable effort had to be performed covering R&D, design, and fabrication and performance evaluations. The energy yield of the fuel increased from initially about 20000 MW·d/t·U to currently as much as 60000 MW·d/t·U, accompanied by an increase of enrichment from initially about 2% to currently close to 5% ^{235}U (PWR). Today combinations of UO_2 with $UO_2 + Gd_2O_3$ fuel are state of the art, and also the use of MOX fuel containing $(U, Pu)O_2$ is increasing with all LWR fuel types.

The core average power densities of commercial power plants developed from around 80 kW/L today up to 108 kW/L for PWRs.

BWR fuel rod arrays with square geometry were starting with 6 × 6 and are now up to 10 × 10. Correspondingly PWR fuel rod arrays with square geometry developed from 14 × 14 and 15 × 15 to now 17 × 17 and 18 × 18.

WWER fuel has a hexagonal geometry with major differences between fuel assemblies for WWER-440 (126 fuel rods) and WWER-1000 (316 fuel rods). However, the fuel rod outer diameter is the same for both fuel assembly types (9.1 mm).

RBMK fuel design is quite different from WWER fuel, due to the fact that the coolant H_2O is different from the moderator (graphite), and the capability to be refuelled online. However, this fuel type is not considered more closely within this documentation.

The details of the fuel rod design were continuously improved as well as the design of the fuel structure components, i.e. the spacer grids, the guide tubes and the upper and the lower tie plate. These developments include the optimization of the materials as well as the thermohydraulic properties of all fuel assembly components and also the neutron-economy of the whole fuel assembly and of the reload and reshuffling strategies for the LWR cores.

During the same time, when impressive developments of the LWR fuel design led to dramatic improvements as well for the specific power generation as for the economy of this generation, the fabrication technology for the needed materials and components of the fuel assemblies was also developed technically and economically.

And finally also the operation reliability improved from the early rather intensive fights with engineering-related problems like rod collapsing and the necessity to change the fuel rod cladding material from stainless steel to Zircaloy to debris-related fuel rod defects and the problems of fretting between spacer and fuel rods. In this way, the fuel rod failure rates could be decreased from 10^{-3} to 10^{-4} in the 1970s to less than 10^{-5} by the end of the 1980s. In the 1990s various unexpected effects led to fuel defects, which interrupted the process of decreasing the failure rate down to 10^{-6} [76]. As an example, Figure 4.9 shows the trend in

fuel failure rates in the USA from 1980 to 2004. Today the best fuel rod failure rates in BWRs and PWR fuel assemblies vary in the range of 2 to 5 × 10^{-6}. Figure 4.10 shows the failure causes in PWR and BWR fuel within the time period 2003 to 2005.

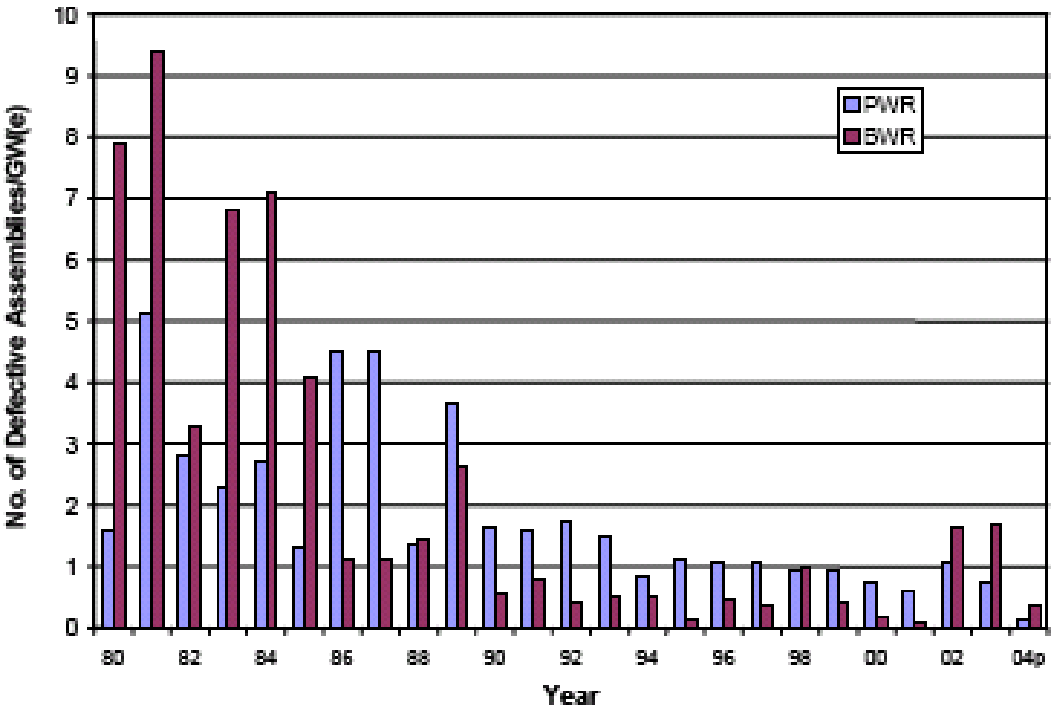
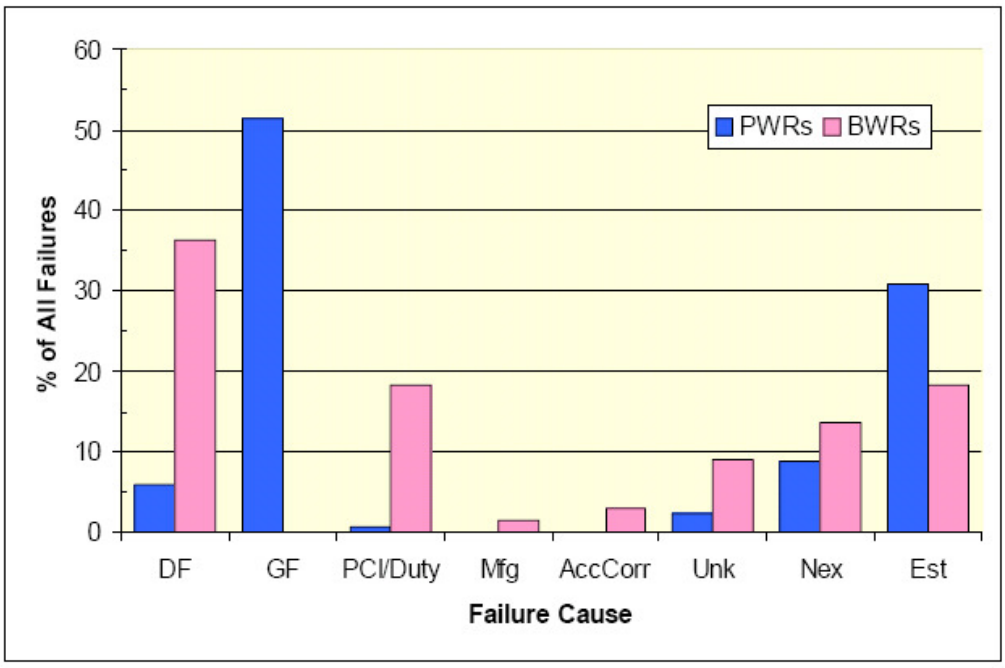


FIG. 4.9. Trend in USA fuel failure rates [77].



DF = debris fretting; GF = grid-to-rod fretting;
PCI/Duty = duty-related; Mfg = manufacturing-related;
AccCorr = accelerated corrosion; Unk = examined, cause indeterminate; Nex = not examined;
Est = estimated based on coolant activity

FIG. 4.10. Failure causes in PWR and BWR fuel during 2003–2005 [78].

A comparable successful development occurred for the WWER fuel during about the same time period covering also design, fabrication and operation. Also here the optimism to reach the magic failure limit of 10^{-6} up to now could not be fulfilled systematically for commercial fuel due to repeatedly occurring unexpected fuel operation problems. On the other hand, the burnup targets could be shifted up to 60 MW·d/t·U. Longer exposure times (5 year cycles) of the fuel assemblies are considered for WWER fuel in a comparable way as for PWR fuel. Also, the fabrication technology for WWER fuel developed successfully during the four decades of its history.

However, it has to be stated clearly that besides the fact that some of the operation problems can be related to the same or similar generic causes like fuel assembly (FA) bow, vibration problems etc., basically there is one very fundamental difference between PWR and WWER fuel performance that leads to different design characteristics typical for these two types of fuel.

It is a well known fact that in PWR fuels the main limiting factor for the liner heat generation rate (LHGR) of the fuel rods and the final burnup of the fuel assembly since more then 2 decades was the corrosion and the corrosion-related hydriding of the FA components made of Zr-based materials. Besides some localized corrosion effects, corrosion and corrosion–related hydriding was never a problem with components of WWER fuel assemblies made of Zr–Nb-based materials. The reason for this difference clearly is also the different water chemistry in western PWRs and WWERs. While in PWRs the pH compensation for the boric acid content in the primary coolant is achieved by LiOH, in WWER cores KOH and NH_4OH is used for this purpose, leading to much lower uniform corrosion of Zr-based materials.

On the other hand, up to today, the design development of WWER fuel assemblies always strongly focused — and still does — on mechanical strength, stiffness and related mechanical properties of the FA components. In turn, this was never a point of specific concern with PWR fuel designs. However, in some of the PWR fuel designs advanced guide tubes have been introduced that contribute to the strength and stiffness of the fuel assemblies.

Finally, it should be mentioned that for BWR and PWR fuel it is state of the art since more than 20 years to be re-constitutable, i.e. they could be repaired after having already been exposed in the reactor.

4.6.1.2. Present situation: International vendors/designers of LWR fuel

After the merging in the western LWR fuel industry mainly two PWR fuel vendors/designers are internationally on the commercial market:

- the AREVA NP (formerly Framatome-ANP) fuel design, compiling the former design developments of Framatome/Fragema and Siemens/KWU;
- the Westinghouse fuel design, compiling the former design developments of the nuclear fuel divisions within the old Westinghouse Electric comp., and the ABB Combustion Engineering Nuclear Operation, for some time under the umbrella of BNFL, and now owned mainly (67%) by the Japanese company Toshiba (and 20% by the US Shaw Group, 10% by Kazatomprom and 3% by the Japanese company IHI).

Both designs are offered worldwide. However, while the AREVA NP design is more focusing on the European situation, the Westinghouse design is more influenced by the demands in the USA. Both companies today design and sell PWR FAs with UO_2 fuel and $UO_2 + Gd_2O_3$ fuel. MOX FAs using $(U, Pu)O_2$ are primarily designed and sold by AREVA NP.

Besides for older plants, today, the international standard for PWR fuel assemblies is a 17 × 17 fuel rod configuration, with the exception of the in Germany operated newer PWR plants built by Siemens which are using an 18 × 18 fuel rod configuration.

With regard to BWR fuel today there are internationally in the market:

- the AREVA NP fuel design, again compiling the former design developments of Framatome/Fragema and Siemens/KWU;
- Global Nuclear Fuels (General electric, USA + international partners), now a joint venture of GE and Hitachi.

Both designers are offering BWR fuel worldwide and design and sell BWR FAs with UO_2 fuel and UO_2 + Gd_2O_3 fuel. Today MOX FAs using (U, Pu)O_2 are primarily designed and sold by AREVA NP (Melox).

Besides for older plants today the international standard for BWR fuel assemblies is a 10 × 10 fuel rod configuration, but also the 9 × 9 configuration still is broadly in use. There are other designers and vendors of BWR/PWR fuel, but just for the domestic demand (e.g. in India, Japan, the Republic of Korea, etc.).

With regard to WWER fuel the vendors' situation did not basically change during the last years. Joint Stock Company TVEL with the support of MINATOM is providing WWER fuel design and manufacturing services to all WWER countries.

Practically, all internationally operating fuel vendors claim that with the present design the fuel assemblies can reach 60 GW·d/t·U. However, in reality the majority of the commercially operated fuel is discharged with burnup of about 40 GW·d/t·U (Refs [79, 80], Fig. 4.11).

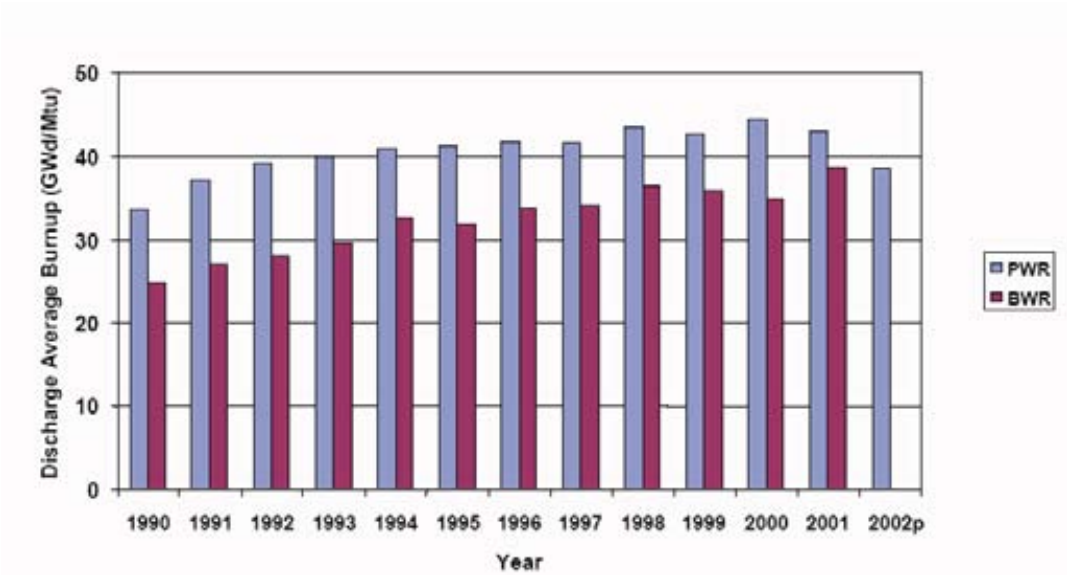


FIG. 4.11. Trends in discharge average burnup — USA.

There is a range of higher FA discharge burnup that extends up to 55–56 GW·d/t·U for AREVA NP fuel (Siemens-design). Similarly, Westinghouse claims [81] to have reached batch discharge burnups of 50 GW·d/t·U.

Along with the increasing burnup the initial ^{235}U enrichment increased as shown in Figure 4.12 for the USA. In fact, in some plants initial enrichments have been realized that go up to the international limit for fabrication and transportation of 5%. With this initial enrichment the discharge burnup can be extended up to 65 GW·d/t·U.

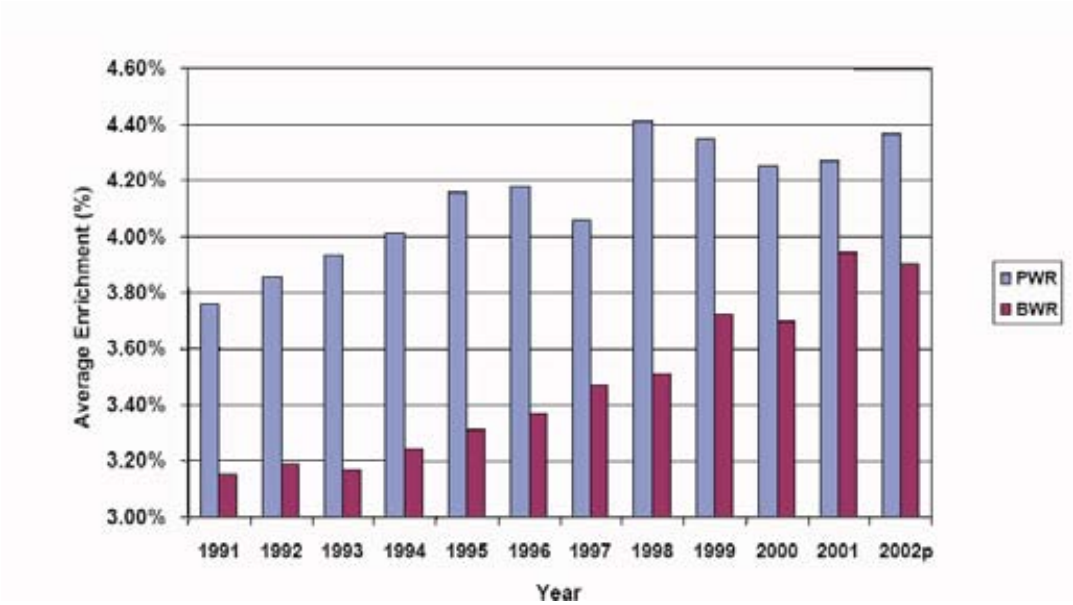


FIG. 4.12. Trends in average initial enrichment — USA [80].

Of course those high discharge burnups can only be realized with an adequately long exposure time as for example 5 one-year cycles. While in Europe the one-year cycle is still the standard case, in the USA, since many years, there is a strong trend to much longer cycles as illustrated in Figure 4.13.

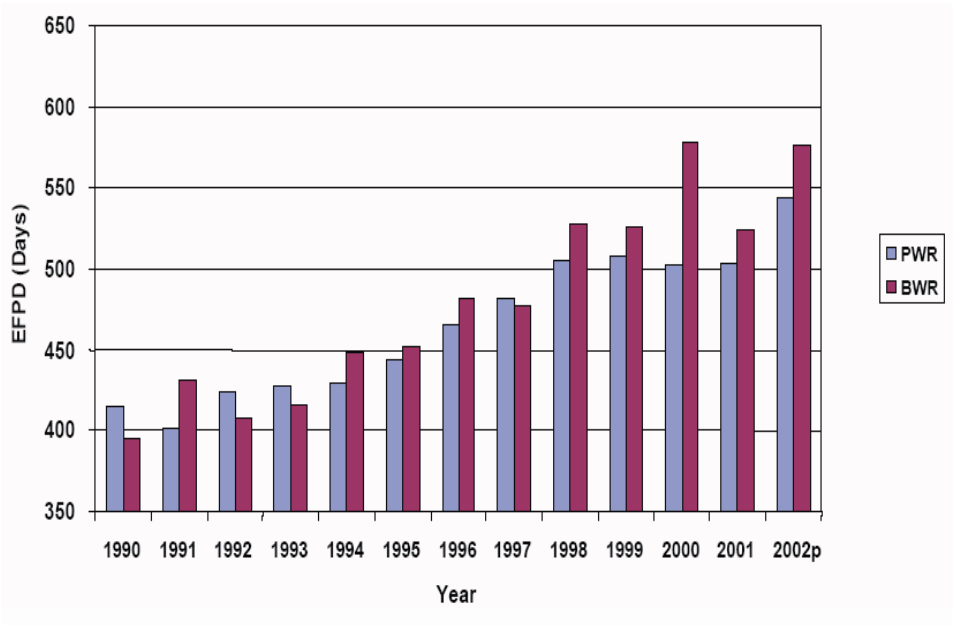


FIG. 4.13. Trends in cycle length — USA.

The discussed increase to a very high burnup has to be accompanied by an adequate development of the core loading strategies. So called Low Leakage Loading Patterns are in use with PWR cores since already many years.

4.6.1.3. Today's operational performance of uranium LWR fuel

As already mentioned in the historical review, up to now western BWR and PWR fuel could not reach the target of a failure rate of 10^{-6} or lower. On the one hand there are still some fuel failures caused by debris fretting. In fact, the anti-debris filters could decrease this problem significantly. However, not all utilities are using fuel with this device. Also the problem of

spacer to cladding fretting could not be fully resolved. In detail, the reasons for this kind of defects are different for different designs.

On the other side, during the 1990s also new types of fuel failures occurred in PWRs that are correlated to a high heat flux in combination with crud deposition on the fuel rod surface. At locations where sub-cooled boiling occurs, i.e. on upper spans of high powered fuel assemblies the deposition is accelerated. Boric acid and LiOH also concentrate in these boiling crud deposits, accumulating boron compounds. The result of this boron buildup in the upper portions of some fuel assemblies is that the core power distribution shifts unexpectedly toward the core inlet (several months after startup of the cycle). This shift is called a crud induced power shift (CIPS), or axial offset anomaly (AOA).

In most cases, CIPs incidents in the USA occurred after the power output of the plant was increased. Under locally thick crud deposits, the temperature at the interface between cladding surface and coolant increases, thus leading to accelerate localized corrosion; as a result fuel rod failures occurred. Consequently more attention has to be paid to localized sub-cooling in advanced high duty PWR fuel designs.

4.6.1.4. Fabrication of uranium fuel for LWRs

In Table 4.10, an overview is given on available data on LWR fuel fabrication facilities worldwide. Altogether there is a large overcapacity (around 50%), which probably will slowly decrease under the ongoing merging and consolidation processes in this part of the nuclear industry.

During the last 15 years the classical way of quality assurance and control was replaced by a new philosophy with new methods to assure the best possible quality of nuclear fuel products. While in the past the reactive efforts were focusing on the detection and removal of quality deficiencies by post-line control of the products after completing fabrication, now pre-planed preventive measures are primarily focusing on process rather than product control. A so-called Total Quality Management (TQM) philosophy was developed and successfully introduced in the fuel production. In this new, mainly process control oriented quality management the statistical evaluation and assessment of processes and product plays an increasingly leading role.

This development still goes on along with continuous efforts to proceed with fabrication automation and computer-based process and process control.

TABLE 4.10. COMMERCIAL LWR URANIUM FUEL FABRICATION PLANTS

Country	Place	Operator	Fuel type	Capacity* (t_{HM}/a)	First operation
Belgium	Dessel	FBFC	PWR	750	1961
Brazil	Resende	INB	PWR	200	1982
China	Yibin	CNNC	PWR	200	1993
France	Romans	AREVA NP	PWR	1200	1979
Germany	Lingen	AREVA NP	BWR, PWR	650	1979
Japan	Tokai-Mura	MNF	PWR	440	1972
	Kumatori-machi	NFI	PWR	284	1972
	Tokai-Mura	NFI	BWR	250	1980
	Kurihama	JNF	BWR	750	1970
Republic of Korea	Taejon	KNFC	PWR	400	1989
Russian Federation	Elektrostal	Mashino Stroitelny	PWR (WWER)	1200	1963
	Novosibirsk	TVEL	PWR (WWER)	400	1949
Spain	Juzbado	ENUSA	BWR, PWR	300	1985
Sweden	Västerås	ABB	BWR	400	1971
UK	Springfields	BNFL	BWR, PWR	860	1996
USA	Columbia	Westinghouse	PWR	1450	1986
	Lynchburg	AREVA NP	PWR	400	1982
	Richland	AREVA NP	BWR	500	1970
	Wilmington	GE Nuclear Energy[21]	BWR	750	1982

* Rod/assembly

4.6.1.5. Status of safety of LWR uranium fuel

Burnup extension of fuel has raised several safety concerns. In particular, the burnup effects on LOCA (loss of coolant accident), and on RIA (reactivity initiated accident) by rod-ejection in PWRs and rod-drop in BWRs, respectively, were intensively discussed during the 1990s. In the case of LOCA, the inclusion of the pre-oxidation level in the cladding oxidation and embrittlement criteria as well as improved core analysis methods has lowered the safety concerns.

For the RIA it was evident from test results (NSRR and CABRI) that the failure threshold at high burnup is significantly lower than the existing licensing limit. Extensive analyses, however, could finally show that (i) the test conditions in general and specifically for the most unfavourable test results are not representative, and (ii) the RIA peak fuel enthalpies for high burnup FA in LWRs are much lower than assumed before and remain below adapted failure threshold values.

Although different approaches were used by the US-NRC and EPRI/ANATECH/EdF, there is now broad agreement that no real danger exists for plant operation. Further experimental confirmation in more representative conditions is planned with a new water loop in CABRI.

[21] Recently, GE formed a joint venture with Hitachi.

Similar conclusions were reached in other countries. In summary, the combined approach of using improved analytical methods and re-evaluating experimental data could relax the situation — independent of the applied methods.

4.6.1.6. Future perspectives for LWR uranium fuel

In the near future, the above-described high burnup targets of up to 65 GW·d/t·U will not be exceeded easily, due to the fact that the internationally established limit of 5% enrichment for fabrication and transportation is reached. However, there are also other reasons for a halt of the development at this burnup figure. On the one hand, some surprises occurred during the last 10 years with regard to the operational behaviour.

Also some traditional operational problems could not really fully be solved up to now; this refers in particular to grid-to-rod fretting failures as the result of vibrations. On the other hand, there are open questions with regard to the safety behaviour of high burnup fuel, e.g. the broadly known question of conservative RIA-limits, and also some so far not yet broadly discussed questions in connection with the LOCA behaviour of high burnup fuel. Therefore, a digesting period is expected as the discharge burnup reaches the range of 65 GW·d/t·U within the next years.

On the other hand, presently there is no technically mature alternative in sight to replace the uranium LWR NFC as the major backbone for the commercial production of nuclear energy.

4.6.2. Mixed oxide fuel for LWR

4.6.2.1. Short history of LWR MOX fuel performance

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 NFC of FR, but due to the slowdown of deployment of FRs 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 is filled with MOX fuel assemblies. Use of MOX in thermal reactors increases the uranium utilization efficiency by a factor of two.

In Europe, in particular in France, in the last decade a considerable amount of experience has been collected with the design, fabrication and the performance of MOX fuel. As already mentioned either Siemens or Framatome/Fragema designed this MOX fuel and the experience is now reported as from AREVA NP fuel [82].

Some MOX lead test assemblies have been designed by BNFL, UK, and were irradiated in Switzerland with interesting data from PIE examination and from ramp test [83]. The overall finding is that the reliability of MOX fuel assemblies is as good as UO_2 fuel assemblies. No rod has ever failed for MOX specific reasons. Moreover, the release rates of fission products into the primary coolant were similar to those observed with defective UO_2 fuel [82].

The burnup of individual MOX FAs has reached 54 GW·d/t_M. In France, a hybride fuel management scheme with 28 UO_2 FAs (3.7% enriched) irradiated for four annual cycles and 16 MOX FAs (Pu content equivalent to 3.25% UO_2) for three annual cycles has been used for all reactors wit MOX fuel since 1994, with load-following conditions since 1995. Recently, the maximum average Pu content was increased from 5.3% to 7.08% to allow the use of Pu coming from reprocessing of highly irradiated fuel. The reactivity of these MOX FAs is equivalent to uranium FAs with an enrichment of 4.3 w/o ^{235}U.

The future perspectives of the French MOX fuel R&D programmes aim for a target burnup of 70 GW·d/t. For this ambitious target it is necessary to improve the current fuel production technology, concerning the (U, Pu)O_2 as well as the design of the fuel rod and assembly structure.

4.6.2.2. Development of LWR MOX fuel in the Russian Federation

In the Russian Federation, historically, MOX fuel was developed in order to use plutonium in fast reactors. Thus, presently, commercial plutonium recycling in thermal reactors is considered unreasonable in accordance with the strategy of nuclear power development in the Russian Federation.

Calculations to study the feasibility of using reactor grade plutonium in WWER-1000 reactors have been performed in the Russian Federation since the late 1980s. Recently, in accordance with the Russian Federation concept of handling plutonium [84], up to 50 tonnes of weapon grade plutonium are to be released as part of the nuclear disarmament process. Under an agreement with the USA, the Russian Federation is to recycle 34 tonnes of weapons grade plutonium as MOX fuel to convert it to a standard spent fuel. The programme may involve all WWER-1000 reactors up to the expiration of their service life, and all WWER-1000 reactors under construction (3 units) — throughout their design service life [85]. From 250 to 280 kg of plutonium may be annually consumed in each of the WWER-1000 reactors. Additionally, Pu fuel could be used in the BOR-60, BN-600 (with a total amount of up to 300 kg of Pu per year) and, prospectively, BN-800 and BREST reactors, and in foreign NPPs licensed for that purpose. Depending on the utilization scope and rate, excessive weapons grade plutonium is expected to be recycled by 2032 to 2050. Most of these studies are performed for developing MOX fuels to convert weapons grade plutonium into the spent fuel standard. For example, work was started in 2001 for the joint experiment of Parallex, as part of which Russian and US pilot assemblies with MOX fuel produced at VNIINM and Los Alamos National Laboratory are being irradiated in a Canadian heavy water research reactor in Chalk River.

As part of cooperation between the European countries, the Russian Federation, and the USA studies are ongoing to upgrade the WWER-1000 reactors for using MOX fuel (produced from weapons Pu). Starting from 1993, the programme of AIDA-MOX 1 was implemented by the Russian Federation jointly with France. It is planned to test pilot FAs with oxide MOX fuel pellets at Balakovo NPP.

It is planned to build a specialized plant at the Siberian Chemical Integrated Works to manufacture oxide MOX fuel out of weapon grade plutonium.

Very important is to build the infrastructure for handling of irradiated MOX assemblies discharged from reactors at NPP sites. It has been shown that spent FAs of a WWER-1000 reactor with MOX fuel (based on weapons grade Pu) with a burnup of 60 MW·d/kg of heavy metal will have a plutonium content of two times and the americium and curium content of three times as much as a uranium assembly with comparable burnup.

4.6.2.3. Fabrication of LWR MOX fuel

A detailed description of MOX production techniques is presented in Ref. [86]. In Western Europe, MOX fuel fabrication for LWR uses the MIMAS process (used by Belgonucléaire, COGEMA) and the Short Binderless Route (SBR, used by BNFL).

In Japan, MOX fuels for core physics and irradiation testing have been manufactured in the JNC plant since 1966. The JNC MOX fabrication plant has been operating since 1972 (10 t_{MOX}/a) for the advanced thermal reactor (ATR) Fugen, and nearly 130 tonnes of fuel

(about 750 fuel assemblies) have been produced till 1999 using the PFPF process, equivalent to about 1.8 t_{HM} of plutonium. A microwave heating method for the co-conversion of plutonium–uranium nitrate to MOX powder was developed by JNC in order to obtain MOX powder directly from the product. Japan (JNFL) decided in November 2000 to build a commercial LWR-MOX fuel fabrication plant with a capacity of 130 t_{HM}/a at Rokkasho-Mura that will be supplied with recycled fuel from the Rokkasho Recycling Plant (RRR) located at the same site. RRR was built by Japan Nuclear Fuel Ltd, and started operation in 2006. The plant's annual reprocessing capacity will be 800 tonnes, compared with about 1000 tonnes of spent nuclear fuel arisings from domestic nuclear power stations annually. Additionally, spent fuel interim storage facilities are expected to commence commercial operation by 2010.

In Table 4.11 an overview is given on commercial and semi-commercial LWR MOX fabrication plants.

TABLE 4.11. COMMERCIAL AND SEMI-COMMERCIAL LWR MOX FUEL FABRICATION PLANTS

Country	Location	Operator	Fuel type	Capacity (t_{HM}/a)	First operation
Belgium	Dessel	Belgonucléaire	LWR	40	1973
France	Cadarache	COGEMA	LWR*	40	1961
	Marcoule – Melox	COGEMA	LWR	1000	1995
UK	Sellafield SMP	BNFL	LWR	120	2005
	Sellafield MDF	BNFL	LWR	8	1993
Japan	Tokai-Mura	JNC	ATR	10	1972
	Rokkasho	JNFL	LWR	130	~2012

* also fuel for FBR fabricated.

India has a domestic plant for fabricating MOX fuel to be loaded in their two BWRs; this plant is operating since 1994 with a capacity of 18 t_{HM}/a.

4.6.2.4. Future perspectives of LWR MOX technology

MOX industry has become a mature business, and parity between UO_2 and MOX fuel prices is requested by the utilities.

Technology developments focuses on improving fuel cycle management by increasing burnup of MOX fuels, the possibility of load-following with MOX fuelled reactors (already demonstrated and licensed in France), and the possibility of an increased MOX fraction in the core loading pattern (100% MOX cores) resulting in a net reduction of the Pu inventory (i.e. minus 15 kg/t_{HM} per cycle in a 900 MW(e) LWR with a 100% MOX core).

4.6.3. Fuel for PHWR

4.6.3.1. History of PHWR fuel

The majority of PHWRs in the world are of the CANDU-type developed in Canada. The core of a CANDU-type reactor consists of horizontal, heavy water cooled pressure tubes, surrounded by a heavy water moderator. A second type of PHWR, the Atucha-type, was designed by Siemens, Germany, and is currently in operation in Argentina. The Atucha reactor is a pressure vessel design, using heavy water as coolant and moderator.

4.6.3.2. Design and performance of CANDU fuel

Currently, operating CANDU-type PHWRs differ in fuel and fuel channel design, and in reactor core size. However, all current Canadian CANDU reactors employ a 100 mm (inner diameter) fuel channel (pressure tube).

In Canada, the reactors at the Pickering station were designed to produce around 540 MW(e), and use a 28-element fuel bundle with 15 mm diameter fuel elements (fuel rods). The larger reactors at the Bruce and Darlington stations were originally designed to produce 900 MW(e) each, using a 37-element fuel bundle with 13 mm diameter fuel elements. The nominal CANDU-6 design is for reactors generating around 700 MW(e). The newest Canadian fuel design is based on a 43-element fuel bundle with two different element diameters (13.5 mm for the inner eight elements and 11.5 mm for the outer 35 elements).

In contrast, India and Pakistan (beginning with RAPS-1 and KANUPP, respectively) based their PHWR design on an earlier Canadian commercial design. These reactors have 82.5 mm diameter fuel channels. The fuel is based on a 19-element bundle (15 mm element diameter).

The CANDU/PHWR fuel bundles are simple and robust assemblies of only 50 cm in length. Particular emphasis is placed on quality control in fabrication, since many more elements (over 30 times) are required per kW·h than in a PWR. By the end of the 1990s, the defect rate for elements was less than 5×10^{-6}. Some of the other countries still had higher failure rates at that time, but are also on the way to lower levels. The CANDU reactor allows defected fuel to be detected, located and replaced without taking the reactor off-line.

In general, most defects in CANDU/PHWR fuel occur in small batches in situations called defect excursions, whereas the failure rates from more statistically scattered defects are usually low. Problems with structural components have been rare. Only one case is reported, where incidental acoustic resonance of the assembly strings with pressure pulsations from the primary circuit pumps lead to severe assembly damages with end plate cracking.

An initial performance problem was stress corrosion cracking (SCC) of the cladding by pellet clad interaction mainly induced by the power ramps during the on-power refuelling, when the bundles are pushed through the channels from low to high power positions. As a remedy the CANLUB fuel was introduced, i.e. a graphite coating on the inner side of the cladding. This feature, in combination with optimised fuelling strategies and operating guidelines, has reduced and partly eliminated the problem. In Canada, there was only one SCC defect excursion during the last 30 years and that one was associated with abnormal operation.

Other defects are mainly caused by either manufacturing problems such as incomplete welds or hydriding, or by debris fretting when small metallic debris from plant construction or maintenance circulates in the fuel channels. A continuous feedback process and national or international improvement programmes within the CANDU community (user group) have significantly reduced both, the frequency of defect excursions as well as the statistical defects.

4.6.3.3. Power generation in CANDU fuel

Though CANDU fuel cycles have been designed to provide increased fuel burnup and improved uranium resource utilization by using slightly enriched uranium, most operators have continued to use natural uranium fuel.

Since the maximum allowable linear element power is the technically limiting parameter and is the same for the 37 and the 28 element bundles, the power density in later Canadian type CANDU (e.g. Bruce station) can be up to 30% higher than in the Pickering station. As the maximum allowable linear element power has been always high, it is also not feasible to

perform significant power uprates without further reduction of the fuel element diameter (Fig. 4.14).

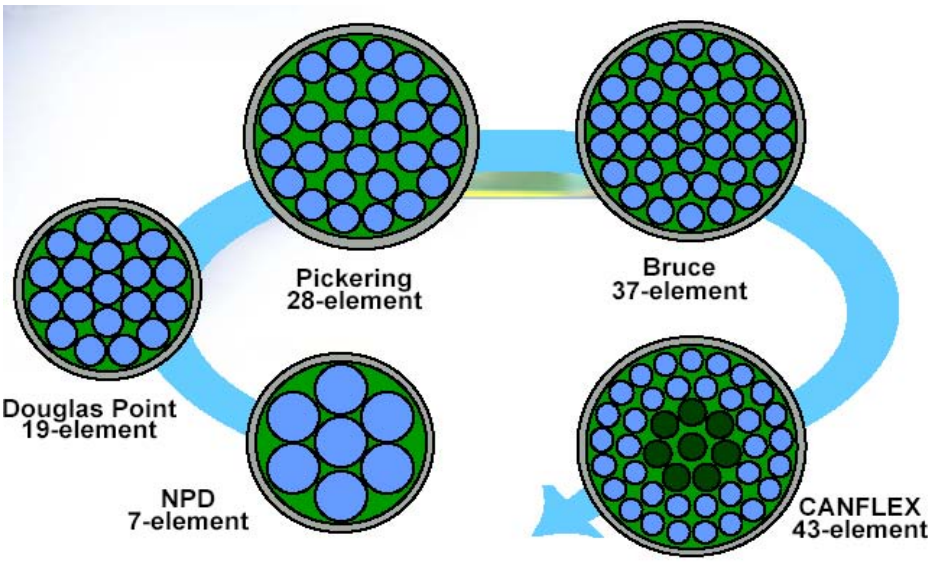


FIG. 4.14. Evolution of CANDU fuel.

The average discharge burnup which can be reached with natural uranium is limited to present values around 8 MW·d/kgU (with some of the CANDU type reactors reaching nearly 10 MW·d/kgU) and cannot be increased without changing the whole fuel cycle by using either slightly enriched uranium (SEU) or other new fuel types.

4.6.3.4. Fabrication of PHWR fuel

In Table 4.12, the commercially operating plants for fabricating PHWR fuel are compiled. The list also contains fabrication capacity for fuel used in a vessel type of PHWR which is deployed in Argentina only.

TABLE 4.12. COMMERCIAL PHWR FUEL FABRICATION PANTS

Country	Place	Operator	Capacity (t_{HM}/a)	First operation
Argentina	Ezeiza	CNEA	160	1982
Canada	Peterborough	GE Canada Inc.	1200	1956
	Port Hope	Zircatec Precision Industries Inc.	1500	1964
China	Baotou	CNNC	200	>2008
India	Hyderabad	DAE	600	1974
Republic of Korea	Taejon	KNFC	400	1987
Romania	Pitesti	SNN	150	1994

4.6.3.5. Perspectives of CANDU fuel

CANFLEX is the CANDU fuel development programme jointly performed by the Republic of Korea and Canada. CANFLEX is a 43-fuel element design of new internal bundle geometry (smaller diameter fuel elements in the outer ring) different from the conventional 37-fuel element bundle. This configuration lowers power peaking factor and thus leads to improved operational margin.

The CANFLEX fuel bundle (Fig. 4.15) design has completed the demonstration irradiations required to qualify it as commercial fuel. Advanced fuel cycles being developed take advantage of the improved thermal-hydraulic margins offered by this design. Fuel cycles under consideration for implementation include the use of slightly enriched uranium, while other cycles, such as the use of MOX fuel, could follow later.

FIG. 4.15. CANFLEX fuel bundle.

An example of such an advanced fuel type, called CANFLEX-LVRF, has completed its qualification process recently. It contains slightly enriched U and provides the ability to uprate reactor power due to its increased safety design margins.

4.6.4. Fuel for fast reactor (FR)

4.6.4.1. Short history of FR fuel

Liquid metal cooled fast reactor technology has been under development for as long as 50 years in many countries, including France, Germany, India, Japan, the Russian Federation, the United Kingdom and the USA (see also Section 2.5.1).

In the USA, a long term development and demonstration programme (till the middle of the 1990s) on metal alloy fuel for sodium cooled fast reactors has been carried at Argonne National Laboratory's EBR-II experimental fast reactor. Pre-industrial scale fabrication technology is well established.

In the Russian Federation the first experimental MOX fuel assemblies for the fast reactors BR-5, BR-10 and BOR-60 were made in the 1970s. Thereafter, from 1980 till 1992, also in BN-350 (with a UO_2 driver core) MOX assemblies were tested. BN-600 operates also on high enriched UO_2 driver fuel but has irradiated many MOX test assemblies since the 1990s. Five basic technologies of producing U/Pu mixed oxide (MOX) for FR fuel have been developed:

1) mechanical mixing of oxides (performed in the Paket facility);
2) sol-gel process (internal and external gelatinization);
3) ammoniac granulation (performed in the Granat facility);
4) carbonaceous co-precipitation;
5) plasmo-chemical conversion of uranium and plutonium solutions into mixed oxides.

Two principal technologies have been developed in the Russian Federation to fabricate fuel rods with mixed uranium–plutonium (mainly from low burnup SNF): pelletising and vibro-compacting of U–Pu powders.

The process Granat has been developed in the All-Russian Institute for Inorganic Materials named after academician A.A. Bochvar (VNIINM). This technology includes granulation following co-precipitation of uranium and plutonium hydroxides with the use of a flocculant characterized by high molecular weight. Co-precipitated uranium and plutonium are converted into oxides to obtain stoichiometric powder. This technology is implemented at PO Mayak (Ozersk).

The technology developed at the All-Russian R&D Institute for Nuclear Reactors (NIIAR) is based on paoelectrochemical granulation of uranium and plutonium under the electrolysis at the cathode in a paographite cell and packing of mixed oxides in fuel elements by means of vibration. The remotely controlled method for fuel production allows using plutonium containing up to 0.5% of americium. The vibro-compacting process directly uses the acid solution from recycling and should have economic merit compared to the classic pelletising route. This technology is implemented at RIAR (Dimitrovgrad).

Vibro-compacting has the following advantages in comparison to pelletising:

- simplicity and reliability of the production process due to the reduced number of process and control operations that makes the automation and remote control of the process easier;
- possibility of usage of the granulate in any form — both in the form of a homogeneous composition and mechanical mixture;
- reduced thermomechanical impact of vibro-packed fuel on the cladding (as compared with a pelletized fuel);
- more flexible requirements for the inner diameter of the fuel rod claddings.

In Japan, JNC has fabricated MOX fuel for the experimental Joyo and prototype Monju fast reactors since 1973.

In France, U/Pu oxide fuel for fast neutron sodium cooled reactors (RAPSODIE) was used starting in the 1960s.

In India, the first experimental fast reactor started up with plutonium carbide fuel in 1985.

4.6.4.2. Fabrication of UO_2 and MOX fuel for FR

Currently, fuel for operating fast reactors contains primarily MOX with a higher Pu content compared to LWR-MOX fuel. Typically, fuel pins are clad in stainless steel and assembled in a hexagonal wrapper tube. The fabrication technology is well established, and FR fuel was supplied successfully in France, the USA, the United Kingdom, the Russian Federation and Japan.

Since fast reactor development so far only reached the level of demonstration plants limited capacities for the fabrication of FR MOX fuel are available, as shown in Table 4.13.

TABLE 4.13. SEMI-COMMERCIAL FR MOX FUEL FABRICATION PLANTS

Country	Place	Operator	Capacity (t_{HM}/a)	First operation
France	Veurey-Voroise	CISN	150	1960
Russian Federation	Elektrostal	Mashino Stroitelny	1	1972
	RIAR	Minatom	1	1965
	Chelyabinsk	Minatom	30	~2010
Japan	Tokai-Mura	JNC	5	1973

4.6.4.3. Carbide and nitride fuel for FR

Mixed uranium plutonium mono-carbide (MC) and mono-nitride (MN) have been identified as advanced LMFBR fuels on the basis of their high heavy metal density, high breeding ratio (and in turn short doubling time), high thermal conductivity resulting in low operating fuel temperatures and excellent chemical compatibility with sodium coolant.

R&D programmes on MC and MN fuels were actively pursued in the USA, France, Germany, the UK and the Russian Federation during the 1960s and the 1970s and a little later in Japan and India. Most of these investigations have been restricted to UC, UN, (U, Pu)C and (U, Pu)N fuels with a maximum plutonium content of 20% and have been very well documented in the monographs entitled Science of Advanced LMFBR Fuels by H. Matzke [87] and Non-oxide Ceramic Nuclear Fuels by H. Blank [88].

However, compared to mixed oxide fuel, the experience on MC and MN fuels is very limited. So far, all over the world, some 1000 kg and 100 kg of MC and MN fuel respectively has been manufactured and irradiation carried out in some 2000 carbide pins and some 200 nitride pins.

In the USA, nearly 700 of helium-bonded and sodium-bonded MC and MN fuel pins have been successfully irradiated in EBR-II and FFTF to high burnups in the range of 10–20 atom%. Most of these pins contained fuel pellets though a limited number of test irradiations were also carried out using vibro-packed fuel pins.

In the Russian Federation, a UC core was in operation in the BR-5 reactor from 1965 to 1971 and achieved a burnup of 6.2 atom%. A large number of UN fuel assemblies were also successfully irradiated in the BR-10 core up to a burnup of 9 atom%. In the BOR-60 reactor too, several UC, U(C, N), (U, Pu)C and (U, Pu)N fuel pins were successfully irradiated to a high burnup.

As part of the Futurix concept [89] irradiation testing of MC fuel pins is carried out in the Phenix reactor in France (and earlier in RAPSODIE/Fortissimo with MC and MN). In the past also in the DFR in the UK, in the BR-2 in Belgium, in the KNK-2 in Germany, and in the JRR-2 and JMTR in Japan such tests were performed.

India is the first country in the world that developed plutonium rich (66% Pu) mixed carbide fuel and used it as driver fuel in their fast breeder test reactor (FBTR). This carbide core is in operation since October 1985 and has so far seen an average burnup more than 100000 MW·d/t without any failure. A second mixed carbide core with somewhat lower plutonium (52% Pu) content is also under irradiation. A few irradiated (25000 MW·d/t) mixed carbide fuel pins have been successfully reprocessed by a wet chemical route for recovery of Pu.

4.6.4.4. Future perspectives of fast reactor fuel

In the USA, a long term development and demonstration programme on metal alloy fuel for fast reactors has been carried at Argonne National Laboratory's EBR-II experimental fast reactor. Pre-industrial scale fabrication technology is well established.

In the Russian Federation, similarly, a long term technology development and demonstration programme on vibro-compaction fabrication for MOX fuel pins for the BOR-60 test reactor at Dimitrovgrad, and pre-industrial scale fabrication technology is well established.

In France, in the fuel development plan (Fig. 4.16) for the gas cooled fast reactor (GFR) an irradiation phase of materials and fuel samples in material testing reactors and available fast reactors prototypes is performed.

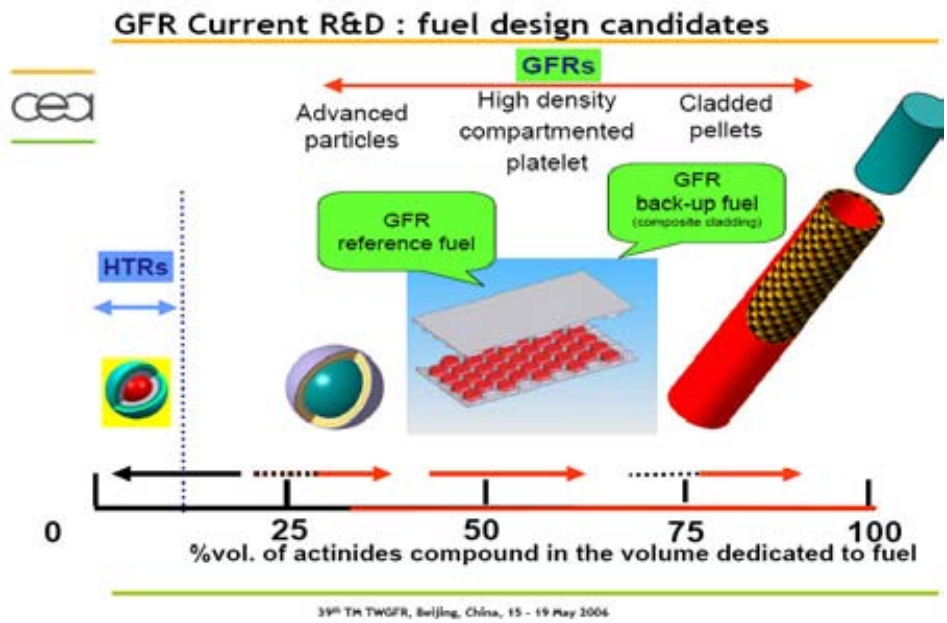


FIG. 4.16. Candidates of fuel design for French GFR (IAEA TWG NFCO 2006).

The development of coated particle nitride or carbide fuel must address several challenging issues that will require fundamental R&D. Specific issues include the development of a coating material for particle fuel as well as for a block-type fuel subassembly that has high temperature resistance. On the other hand, since the helium gas cooled reactor was selected as one of the candidate reactor types at the GIF project, it may be possible to achieve a break through regarding these fundamental issues by international cooperation.

In Japan, metal fuel for the sodium cooled fast breeder reactor is being developed. Core design with such a fuel has been performed (Ref. [90]), advanced cladding materials are investigated, e.g. oxide dispersion strengthened ferritic and precipitation hardened steels (Ref. [91]), and innovative fabrication techniques (Ref. [92]) are studied. Innovative fuel fabrication technologies (Fig. 4.17) include simplified pellet, sphere/vibro packing, and metal casting processes.

Reprocessing	Simplified PUREX Process	Alternative Aqueous Process	Oxide-Electrowinning Process	Metal-Electrorefining Process
	Dissolution	Direct Extraction (Supercritical Fluid Extraction TBP-CO_2-HNO_3)	U Electrowinning	Oxide Reduction
U Recovery	Crystallization			U Electrorefining (solid cathode)
U, Pu,MA Recovery	Single Cycle Extraction		MOX Electro-codeposition	U, Pu, MA Electrorefining (Liq. Cd cathode)
MA Recovery	SETFICS Process /TRUEX Process	Ion Exchange / Amine Extraction	MA Electrowinning	Cd Extraction (pyro contactor)

Oxidation

Fuel Refabrication	Pellet Short Process	Sphere Packing Process	Vibro-Packing Process	Casting Process
	Simplified Pelletizing	Gelation (MOX−MA)	Granulation (MOX, MN−MA)	Casting (U−Pu−Zr−MA)
	Stacking	Vibration Packing	Vibration Packing	Stacking
	MOX Fuel		MOX Fuel	Metal Fuel

: MOX Aqueous Recycle : MOX Non-Aqueous Recycle : Metal Non-Aqueous Recycle

FIG. 4.17. Innovative fuel fabrication technologies for FR in Japan.

4.6.5. Fuel for HTGR

4.6.5.1. History of HTGR fuel

The situation for HTGR fuel is somewhat similar to thorium fuel. HTGR fuel was developed and exclusively used in the past in demonstration plants (see Section 2.4.1 and 3.4.2).

Very good experience was collected with the coated particle concept that allows rather high coolant temperatures. The coated particles were small kernels (~few hundred microns diameter) of e.g. uranium oxide-carbides, with the ^{235}U enriched up to 9%. These kernels were coated by paolytic carbon and silicon carbide, providing containment for the fission products for temperatures up to 2000°C.

They were deployed in two fuel element /assembly designs (see Section 2.4.1):

- the pebble bed design, using coated particles in a ball shaped graphite shell (German experience);
- the prismatic graphite block design, which was used in the Fort St. Vrain high temperature gas cooled reactor (US experience).

These two variants of fuel arrangements are still of current interest, each containing about 15000 particles. Both designs provide a high level of inherent safety, including strong negative temperature coefficients on reactivity.

4.6.5.2. Fabrication of HTGR fuel

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). A process based on sol-gel microsphere pelletization has been developed in Germany for fabrication of high density oxide pellets for HTGRs, at relatively low compaction pressure and low sintering temperature, avoiding dust generation.

4.6.5.3. Future perspectives of HTGR fuel

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. As long as uranium is used as oxide experience accumulated in Germany and in the USA could help to drive this development to commercial maturity within the next decade.

Several R&D programmes are underway worldwide to establish fabrication processes of HTGR fuel. In the Republic of Korea, as part of the development programme for the VHTR (GEN IV), a TRISO type fuel is being investigated experimentally (e.g. kernel fabrication, ZrC coating, irradiation tests) and analytically (e.g. computer codes, material database). Another example is the French development programme of fuel for the GFR (Fig. 4.16). In the USA, in the advanced test reactor (ATR) TRISO particle fuel with ZrC is also been developed [93] and will be irradiated [94] as part of the advanced gas reactor development programme.

4.6.6. Fuel for Magnox reactors

Presently, there is still the Magnox-type fuel (natural U metal) in commercial use, but only in the UK. Since it is assumed that this fuel type will not be developed further, it is not discussed in more detail here[22].

4.6.7. Future advanced fuel designs

There are two fuel types with interesting potential for future technological applications in thermal and/or fast reactors:

- Low strain resistant fuel (LSRF);
- Inert matrix fuel (IMF).

The two concepts are shortly discussed in the following two sections.

4.6.7.1. Low strain resistant fuel (LSRF)

Basically it is an old idea to mix additives to the UO_2 to decrease creep resistance of UO_2 pellet fuel and thus minimizing mechanical interaction between the fuel pellets and the cladding tube during power ramping. However, in the past respective development effort in various countries did not come up with attractive properties of pellets.

Recently, results from Russian development programmes show promising behaviour data [95]. Under laboratory conditions, at different temperatures, the influence of various additives, i.e. iron oxides, niobium, silicon, aluminium and their combinations, on creep behaviour of UO_2 was studied. The subsequent comparative analysis proved that the most promising approach to decreasing strain resistance of UO_2 pellets is to alloy uranium dioxide with mullite ($3Al_2O_32SiO_2$) and niobium oxide. This forms intergranular precipitates of low shear resistance in the pellet. The phase and chemical analysis revealed that niobium oxide forms a solid solution, while mullite is located along the grain boundaries.

The most important characteristics that govern the behaviour of claddings under stress, the life-time parameters and state of a fuel rod in transients and accidents are properties of fuel pellets determined in in-pile experiments: irradiation induced creep, swelling, densification. In this Russian programme [95] specimens of modified UO_2 (containing additives) were investigated inpile. The mass contents of alloying elements were as follows: Al-(0.1 to 0.13)%, Si-(0.06 to 0.075)%, Nb-(0.06 to 0.1)%. The irradiation of this fuel was carried out in the IVV-2 reactor at a fission density of ~(1/5 to 1/7)/10^{13} fiss/cm^3s to determine its irradiation induced creep, densification, swelling and stresses suppressing swelling.

From these Russian investigations it was concluded that based on the acquired data that the use of modified uranium dioxide should significantly decrease the fuel-cladding mechanical interaction.

4.6.7.2. Inert matrix fuel (IMF)

Since the end of the 1950s, many countries have studied the features of nuclear fuel with an inert matrix. A good overview on this fuel concept is provided in Ref. [96]. The basic idea is

[22] There maybe an interest again in the future due to the development of metal fuel for advanced fast reactors.

to use composite fuel with fissile material dispersed in a metallic or ceramic matrix. Due to its increased thermal conductivity this leads to significant reductions of temperatures in the fuel during operation. As a consequence the fission gas release is strongly reduced, because the release is strongly dependent on fuel temperature. In addition, these composite fuels include the concept of a double barrier (matrix + cladding) to fission gas release outside of the fuel rod.

IMF has two compositional options: homogeneous materials, which are solid solutions such as oxide, nitride or metal (alloy), and heterogeneous materials, which are composites of cercer, cermet or metmet. These materials may be used as cylindrical pellets, prismatic designed blocks, or microspheres in the form of sphere packs or kernels.

In an IAEA Technical Committee Meeting held in Moscow in 1996 [97] an international exchange of experience with this fuel concept was performed. In particular, the experience from work in Argentina, France and the Russian Federation was presented and discussed. It was shown that such so-called cold fuel might:

- Allow to raise the burnup to 100 $MW \cdot d/kg_{HM}$ by using a Pu fraction instead of high ^{235}U enrichment.
- Increase the margin for reactor operation flexibility.
- Enhance safety margins in case of LOCA or RIA accidents.

It was clear at that time, that much further research and development work would be needed to end up with commercially mature solutions.

A survey is available on the status of this development in the Russian Federation with regard to LWR application. Since the 1990s, inert matrix fuel (IMF) with a once-through strategy was adopted [98] considering the following properties:

- neutron properties, i.e. low absorption cross-section, optimal constant reactivity, suitable Doppler coefficient;
- phase stability, chemical inertness and compatibility;
- acceptable thermo-physical properties, i.e. heat capacity, thermal conductivity;
- good behaviour under irradiation, i.e. phase stability, minimum swelling, retention of fission products and residual actinides;
- in case of a once-through strategy, high insolubility of fission products. If instead a multi-recycling strategy would be applied, good leaching properties for reprocessing are required.

In the Russian programme a zirconium-based alloy was used as matrix material [99]. Fuel particles as uranium dioxide microspheres with about 500-µm diameters are placed in the matrix, and they take more than 70% of the whole volume.

Based on this Russian research, firstly, it is possible to rule out or, at least, to reduce significantly the fuel rod plenum for gaseous fission products. Secondly, the probability for fuel rods to loose their leak-tightness significantly decreases, and, therefore, the IMF rod can be made as a bulk rod rather than a pellet column, which will result in enhanced rigidity and geometric stability of the fuel. These features may improve the resistance against vibration of the fuel assembly and thus improve the operating conditions of spacer grids.

Also matrix fuel elements may be operated in load-following modes reducing existing limitations of number, speed and size of load cycles. Accident analyses performed for reactors of WWER-1000 type with an inert matrix fuel core have shown that in case of LOCA the maximum fuel temperature would not exceed 500°C, thus leading to significantly improved cladding failure behaviour.

Experiments performed in Russian WWER-1000 reactors under power operation modes reached a burnup of 60 MW·d/kgU, and all tested fuel elements retained tightness and geometrical dimensions. Preliminary calculations and measurements show a burnup potential of at least up to 120 MW·d/kgU.

Reprocessing technology of irradiated matrix fuel is less developed. Today, there are only preliminary feasibility studies available for matrix elements treatment using thermal gas–chemical processing in the open air at a temperature of 1000–1100°C during several hours.

CHAPTER 5.
BACK END OF NFC

In the following sections the history, current status and perspectives of the back end of NFC is laid out covering the areas of SNF management, reprocessing and recycling, and HLW management including partitioning and transmutation.

5.1. Spent nuclear fuel management

With the growth of industrial deployment of nuclear energy spent nuclear fuel (SNF) has become a subject of major consideration in the nuclear industry. SNF management is an integral part of the nuclear fuel cycle. It includes the technical operations beginning with the discharge of SNF from a power reactor and ending currently either:

- with the reprocessing of SNF and recycling of plutonium and uranium in new mixed oxide fuel (closed cycle) and the final disposal of the (high level) waste resulting from reprocessing, i.e. the fission products and also fissile material that is no longer recycled;
- with the direct disposal of the SNF elements (open, once-through cycle).

Intermediate and low level waste results from any type of fuel cycle, regardless if it is a closed or an open cycle. For example, a typical 1000 MW(e) LWR produces about 200 to 350 m^3 of low[23] and intermediate[24] level waste.

Another position is currently taken by a majority of countries, i.e. deferring the decision between the above mentioned two options, which may be called a 'wait and see' strategy with interim storage, which provides the ability to monitor the storage continuously and to retrieve the SNF later for either reprocessing or direct disposal.

5.1.1. Worldwide arisings of spent nuclear fuel

In this context it has to be noted that the worldwide growth of generation of electric power using nuclear energy is continuing. While the rate of growth has essentially levelled out in Europe and North America, it has increased and is still increasing significantly in Asia. These trends have impact on SNF management in the various regions. The worldwide SNF production continues at about 10000 t_{HM}/a.

The trend to higher fuel burnup potentially reduces this amount. However, this requires higher enrichment of the fresh fuel, and the use of plutonium in mixed oxide fuel, which leads to other SNF characteristics (i.e. a higher decay heat decreasing more slowly over time). This requires a longer interim storage period (before final disposal) than for the present SNF with burnup lower than 40 GW·d/t.

In many countries, the lack of final repositories and the deferral of the decision will lead automatically to long storage periods even of uncertain duration. The lifetime of existing storage facilities will have to be extended and new facilities for long term storage have to be built.

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[23] 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.

[24] 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.

About 35% of the SNF arising came from each of the two regions West Europe and North and South America and 15% of each of the two regions East Europe and Asia and Africa [100]. Figure 5.1 shows the past and projected regional SNF arisings in t·U/a.

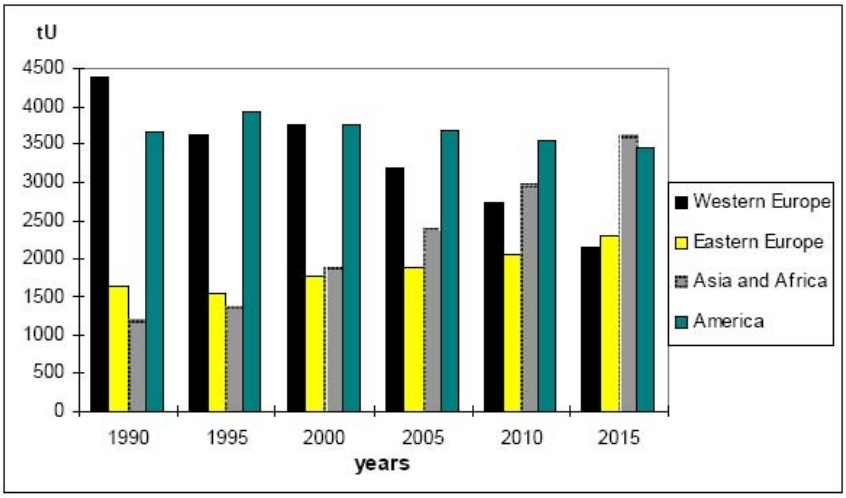

FIG. 5.1. Annual spent nuclear fuel arising in world regions.

End of 2005 [5], the total amount of SNF accumulated worldwide was about 276000 t_{HM}. About 30% of this fuel was reprocessed. The remaining SNF is stored in at reactor (AR) pools and away from reactor (AFR) wet and dry storage facilities.

In Figure 5.2 it can be seen that the fraction of SNF being stored relative to the fraction to be reprocessed is expected to increase.

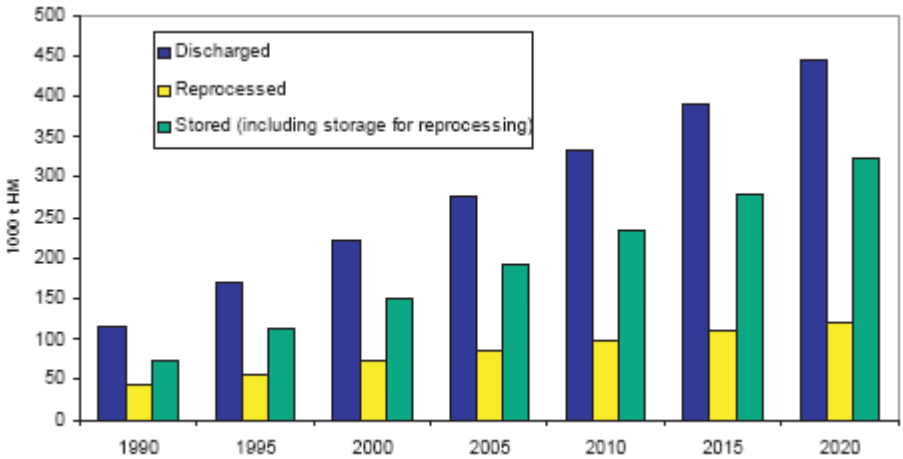

FIG. 5.2. World spent nuclear fuel arising and amount of spent nuclear fuel reprocessed and stored [5].

Projections indicate that the cumulative amount generated in the world by the year 2020 may surpass 440000 t_{HM} of which about 320000 will be in storage. Of this total amount in 2020, the amount in western Europe will remain about the same (because of reprocessing SNF) and will be around four-fold in Asia and Africa.

On a regional basis, the picture looks different. About 50% is stored in North and South America (because there is no reprocessing), 25% in West Europe and the remaining part in East Europe and Asia and Africa.

Table 5.1 provides an overview on the capacity of national facilities for storage of spent nuclear fuel (SNF).

TABLE 5.1. CAPACITY OF NATIONAL STORAGE FACILITIES OF SNF AT REACTOR SITE (AR) AND AWAY FROM REACTOR SITE (AFR) (STATUS 2004, REFS [5] AND [49])

Country	SNF storage (t_{HM})	Country	SNF storage (t_{HM})
Argentina	6500	Armenia	88
Belgium	2000	Brazil	318
Bulgaria	1156	Canada	35271
China	420	Czech Republic	936
Finland	2115	France	25762
Germany	14920	Hungary	500
India	3515	Japan	11521
Kazakhstan	30	Republic of Korea	8738
Lithuania	2101	Mexico	984
Netherlands	86	Pakistan	160
Romania	940	Russian Federation	20928
Slovakia	1050	Slovenia	361
South Africa	670	Spain	3820
Sweden	6500	Switzerland	904
Ukraine	5150	UK	12000
USA	62000		

Generally speaking, many countries with a small nuclear power programme or only research reactors will soon face serious problems of needing extended interim storage and disposal of their SNF. However, the high specific costs for the construction of away from reactor extended interim storage facilities and/or geological repositories in such countries is obviously not reasonable and, therefore, from an economical point of view, access to a regional/multinational interim storage facility and/or repository for their fuel will become more and more interesting. Safety, safeguarding and economic advantages clearly support the deployment of regional/multinational SNF storage facilities. However, there are still various problems to be solved as to find operators of such facilities with governmental support and to convince countries with proliferation concerns to participate.

5.1.2. National strategies for management of spent nuclear fuel

As already shortly outlined in Chapter 3.2.1, presently, SNF management is totally a national task and effort everywhere in the nuclear world.

Countries like Canada, Sweden, Finland and the USA[25] decided to retain their plutonium in the SNF and are planning to put their fuel in long term interim storage followed by final disposal in deep geological formations.

At least, part of the fuel was reprocessed in the past from countries like Belgium, Germany, and the Netherlands. France, India, Japan, the Russian Federation, Switzerland and the UK continue to reprocess part of their SNF. Countries with smaller nuclear programmes are currently deferring their decision on which of the strategies to select and are storing their fuel.

[25] The USA is currently reconsidering their strategy regarding SNF (see Section 2.6.2, GNEP).

Generally, there is international agreement to finally dispose of HLW (SNF or waste from reprocessing) in deep geological repositories with stable structures, e.g. granite, crystalline rock, clay, salt, etc. However, it has to be noted that to date there is no practical urgent need for final HLW deep geological depositories, as surface storage of SNF for 40 to 50 years reduces heat and radioactivity of SNF by a factor of about 1000 which makes handling and storage much easier.

Thus, currently, intermediate or interim (long term) storage of HLW is being practiced in all nuclear countries as outlined in the following section.

5.1.3. Interim storage of spent nuclear fuel

The length and method of interim storage has been a topic of many discussions within the utility industry, with regulatory bodies and with international organizations like the IAEA because of economic, licensing and societal issues. Also, an increasing number of countries consider SNF as a valuable energy resource that should be maintained for the future. Thus, also the aspects of sustainability became part of the discussions.

In several countries, storage facilities have been initially licensed to operate for periods up to 50 years. More and more periods up to 100 years or longer are now under consideration. And finally, delays in decisions on final disposal options have increased the demand for SNF storage capacity and this trend is expected to continue.

Historically one did distinguish between storage of SNF at the site of the NPP (at reactor — AR) and storage in special facilities away from the NPPs (away from reactor—AFR). Originally the AR storage just was buffer storage for the power plant for better disposition of SNF. In the meanwhile, often it had to be used to supplement not available or not sufficiently available AFR storage. In some countries, like in Germany, it became official policy of the political administration to force the utility to build long term intermediate storage capacity on-site.

An intermediate storage is always necessary before SNF can be either sent to reprocessing (closed fuel cycle) or to processing for direct final disposal (open fuel cycle). As stated above, this intermediate storage is necessary to get time for cooling of the SNF. However, due to the delay of final depository solutions practically everywhere in the world, nowadays long term intermediate storage de facto is replacing the not yet available final depositories.

Today there are two technologies in use for intermediate storage:

- wet storage, where the SNF is disposed in pools containing circulating water for cooling;
- dry storage, where the cooling of the SNF is provided by ventilation or natural convection.

An increasing use of higher enrichment is accompanied with higher burnup, and also the use of mixed oxide (MOX) fuel results in higher decay heat levels from these fuels. Therefore, wet storage will remain the preferred approach for interim storage during the first decade after discharge. After sufficient decay of fission products and, especially, where long term storage is foreseen (up to and beyond 100 years in some Member States) dry storage under inert conditions or in air becomes the preferred option.

Short term wet storage has become a classical technology that does not need any further discussion. Long term wet storage is not further discussed here due to its low future perspectives. However, more details on the technical situation in this area is available in an IAEA evaluation [101]. The trend unambiguously goes to dry storage everywhere in the world.

Technically the increasing storage times request answers to three questions:

- What will be the behaviour of the SNF over the full time of storage?
- What will be the behaviour of the (wet or dry) storage facilities, including necessary maintenance, surveillance, etc.?
- How can the regulatory concerns be addressed for storage times of fifty to hundred years, and sometimes even more (France)?

Thus, the prediction of the integrity and retrievability of SNF are the main discussion topics for SNF behaviour regardless of the storage system and time period envisaged.

In order to be able to assess the long term integrity of the SNF the mechanisms that might degrade the fuel and fuel structure have to be known. Cask storage, in comparison to all other storage techniques, presents the greatest challenge to long term fuel cladding performance, as a result of the high initial operating temperatures during the early years of storage. Since the wet storage will more and more disappear in the future, all considerations focus on the SNF behaviour under dry storage conditions.

An IAEA coordinated research programme performed on the long term SNF storage from 1994 to 1997, and subsequent evaluations of this programme [101] came to the following conclusions:

- The creep strain capability has to be known for all materials used. Creep tests should be done under realistic conditions (irradiation, oxidation, hydrogen, local effects, etc.) to assess the straining capability.
- A representative creep law is needed. Dependent on the margins, irradiated or non-irradiated material can be used.

Following these conclusions it was recommended in Ref. [101]: "To assess the long term effects, for example the interaction of different materials and low temperature gas release, an international surveillance program on the spent fuel behavior under dry storage conditions is proposed. This program should monitor long term behavior to assure that no unexpected phenomena occur and to verify the predicted behavior."

In the above mentioned IAEA evaluation [101] it became clear that in case of long term (interim) storage for periods up to 100 years and beyond the scientific, technical and regulatory communities have to extrapolate existing views and practices to safely operate and control the nuclear energy cycle.

Since this long term storage is also not deployed to replace the final deposition, the storage concept has to provide the capability to safely re-handle the SNF at any point in time after initial storage. This leads to the following requirements on any long term storage system:

- Enough relevant technical knowledge must be secured throughout the period of time envisaged for the operation.
- Answers have to be found how to maintain this know-how.
- Funding will also have to be secured.
- Sufficient general industrial infrastructure must be kept operational, i.e. relevant data and information describing the storage system, including SNF packages stored in it, must be kept in a secured way, to be readable and usable much later.

With these considerations the following key design principles have to be observed:

- The destination of SNF at the end of the storage period (options should be unknowingly foreclosed: geological disposal, reprocessing, others) should be known, if possible, and considered in the storage system design.

- Retrievability is a key function for long term storage and a safety goal.
- The storage design must incorporate a certain degree of flexibility to adapt to future needs and conditions.
- Safeguarding the storage facility for long periods of time is a prerequisite.
- Total cost to construct and operate the facility must be affordable now and in the future.
- The operating system must make use of as many passive components as practicable during the storage phase.
- Maintenance needs have to be evaluated and considered at the design stage.
- The overall environmental assessment must lead to a prediction of cumulative impacts over time, which meet or exceed current standards and complies with the ALARA principle.
- The long term storage facility should be publicly acceptable.

In Table 5.2 long term storage conditions and the present status of knowledge is compiled based on the above-mentioned IAEA evaluation [101].

TABLE 5.2. STORAGE CONDITIONS

Item	Known	Partly known	Unknown
Safety case	x		
Duration		x	
Heat removal		x	
Temperatures		x	
Monitoring		x	
Radiation protection	x		
Horizontal or vertical storage	x		
System type (cask, silo, vault)		x	
Indoors or outdoors	x		
Handling systems		x	
Modelling measurement		x	
Contingency accident/incident	x		
Export/import facility	x		
Inspection/repack		x	
Size configuration		x	
Contingency measures		x	
Decommissioning + activation of mat.		x	
Waste arising + costs		x	
Programme-endpoints-review-decision point		x	

It can be concluded that the today envisaged long term interim storage schedules create substantial technical problems, as well as regulatory concerns, to be solved in not to far a future.

5.2. Reprocessing and recycling

5.2.1. Introduction

There is a considerably broad technical experience available from commercially reprocessing U/Pu oxide fuel since several decades. But, there is limited experience with the recycling of other fuel types, mostly metallic U/Pu fuel, and limited access to this experience. The

available experience confirms the technical and economical feasibility of large scale reprocessing of U/Pu oxide fuel, and proves the technical feasibility of reprocessing some other fuel types, in particular metallic U/Pu fuel.

However, this experience also shows that reprocessing is one of the most sensitive steps of the whole fuel cycle with regard to:

- proliferation resistance;
- protection of people and environment (against chemical and radiation hazards).

These concerns are among the reasons why large scale commercial reprocessing is performed only in very few countries, like in France and in the UK, and reprocessing is performed there for many other countries. And there are a lot of debates on these concerns inside and outside of these two countries [102]. The technical reasons for these concerns are the very high amount of fissile and of radioactive materials to be handled and the risk of criticality in a sequence of thermo-chemical processes.

Therefore, any discussion on the present status and the future perspectives of this technology has to be performed considering the above-mentioned concerns.

On the other hand, this technology is key for a substantial progress of the NFC towards improved economics, saving of raw material recourses, and strengthening proliferation resistance.

A good overview on the status and trends in SNF reprocessing is provided in Ref. [103].

The worldwide capacity of reprocessing amounts to about 5600 t_{HM}/a and represents a cumulative experience of about 80000 t_{HM} processed during over 50 years.

Table 5.3 provides an overview on national capacities of reprocessing.

TABLE 5.3. CAPACITY OF NATIONAL REPROCESSING FACILITIES (REFS [5, 49])

Country	Location	Capacity (t_{HM}/a)	First operation
China	Lanzhou	100	>2010
France	La Hague	1700	1967
India	Kalpakkam Tarapur	160 100	1998 1974
Japan	Rokkasho	800	2007
Russian Federation	Mayak	400	1977
UK	Sellafield	2400	1967

5.2.2. Reprocessing technologies

Currently, the U/Pu oxide fuels used worldwide in thermal power reactor systems today are commercially reprocessed by aqueous techniques; these systems, using primarily the PUREX (Plutonium and Uranium Recovery by Extraction) process, are fully developed and deployed commercially for full fissile (plutonium) recycling. The PUREX process is the most commonly used aqueous method to separate U and Pu from SNF containing natural, low or high enriched uranium, and is the reference process for LWR/UO_2 and LWR/MOX reprocessing.

The main disadvantage of the PUREX process is the generation of pure plutonium and its disability to separate individual minor actinides (MA e.g. neptunium, curium, and americium) and specific long living fission products (e.g. ^{137}Cs, ^{90}Sr, ^{99}Tc, ^{14}C, ^{129}I).

Thus, during the last decades substantial R&D has been performed to recover minor actinides (and fission products) in advanced aqueous processing schemes; examples of such processes based on PUREX are COEX (France), NUEX (UK), THOREX (India), NEXT (Japan), and REPA (Russian Federation).

The European research programmes are also directed to different options in the PUREX process to achieve the partitioning of MA:

- the PUREX process adapted for Np recovery;
- the European Extended PUREX process for MA recovery with its three alternative approaches.

In France also an advanced SNF separation process called GANEX [104] is under development (Fig. 5.3).

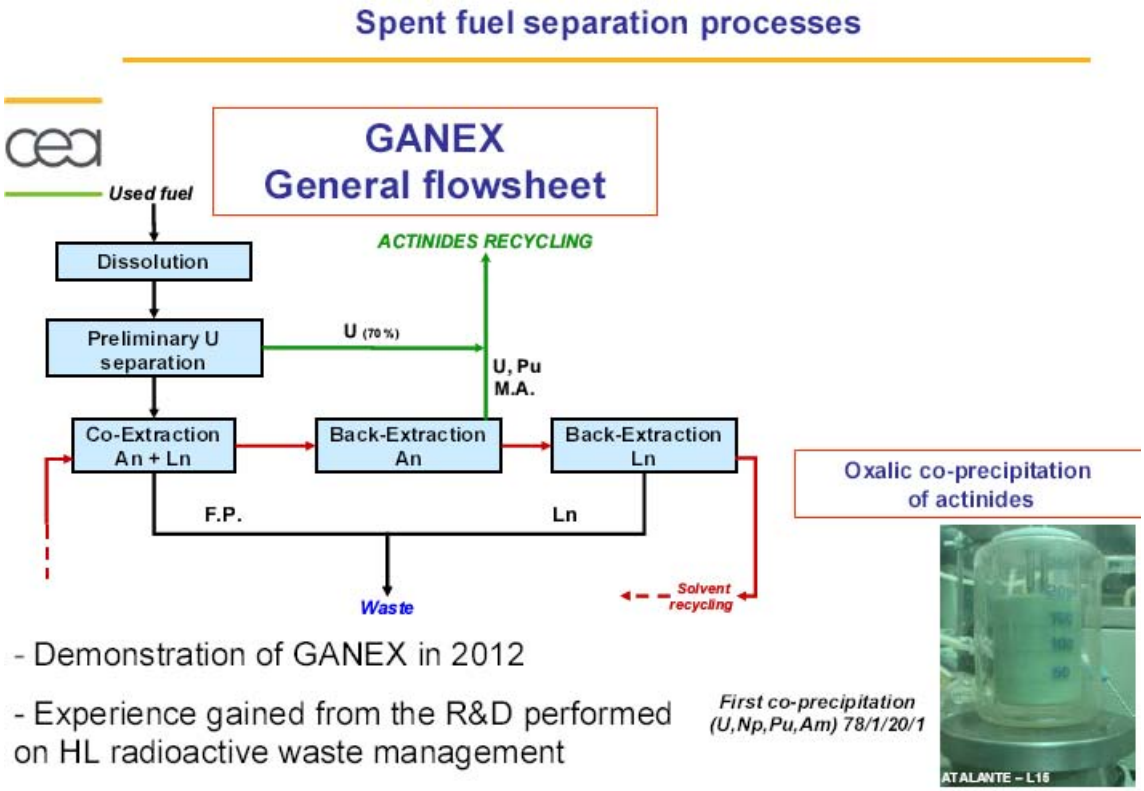

FIG. 5.3. French development of spent nuclear fuel separation processes.

GANEX is called a homogeneous recycling process. Its first demonstration is planned for 2008 in the ATLANTE facility [105].

In Japan, different institutions are studying several alternative techniques. The (JAEA) PARC process [106] for LWR fuel cycle including RMWR is an advanced PUREX process having partitioning functions of MAs and long lived fission products (LLFP, such as ^{14}C, ^{129}I, and ^{99}Tc) as shown in Figure 5.4 (Refs [103, 107]). JAEA also has been developing the Four Group Partitioning Process for recovery of MA and selected fission products (^{137}Cs, ^{90}Sr, ^{99}Tc and Pt) from HLW [108]. JNC has been developing an Advanced Aqueous Process (NEXT Process) for recovery of MA for a FR fuel cycle [109]. To minimize the ecological burdens a new R&D strategy, called advanced optimization by recycling instructive elements (Adv.-ORIENT) was initiated recently [110]. It aims at separating MA, lanthanides and most of the long lived fission products to be transmuted afterwards.

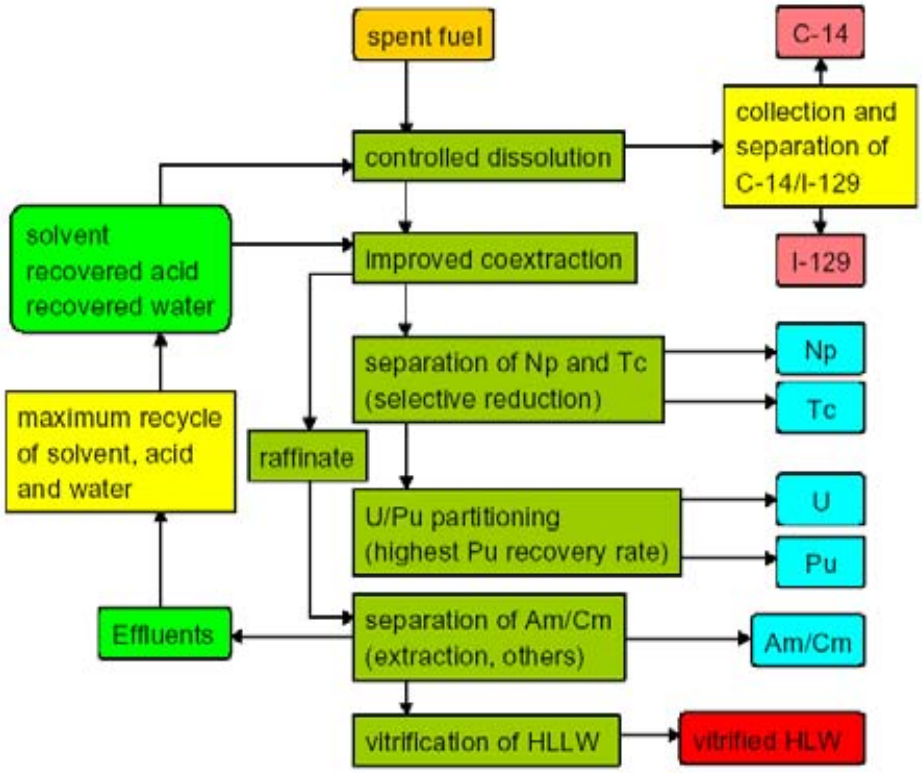

FIG. 5.4. JAEA PARC process for recovery of MAs and long lived fission products.

In the USA, within the Advanced Fuel Cycle Initiative (AFCI, see also Section 3.4.4) programme, the UREX process (Fig. 5.5) has been developed which can reduce the volume of high level waste volume by removing uranium, TRU and high heat generating fission products [111].

These advanced processes (Fig. 5.6) all allow selected extraction of various MA and specific LLFP from solutions of dissolved SNF, but also represent attempts to advance the PUREX standard to improve economics of reprocessing and provide reductions in secondary waste streams.

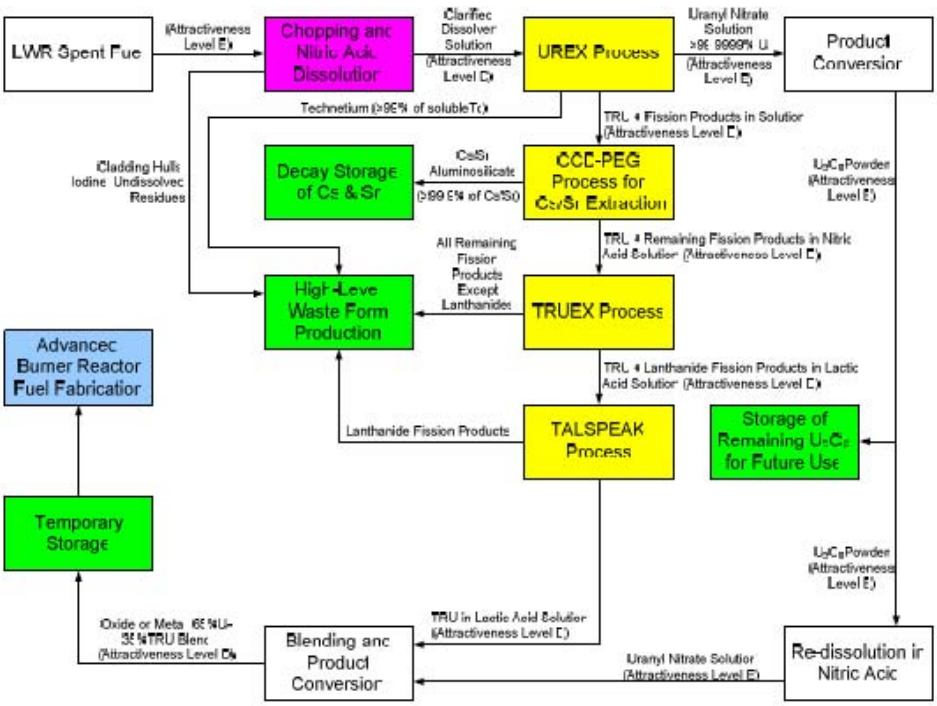

FIG. 5.5. US concept of spent nuclear fuel treatment (IAEA TWG NFCO 2006).

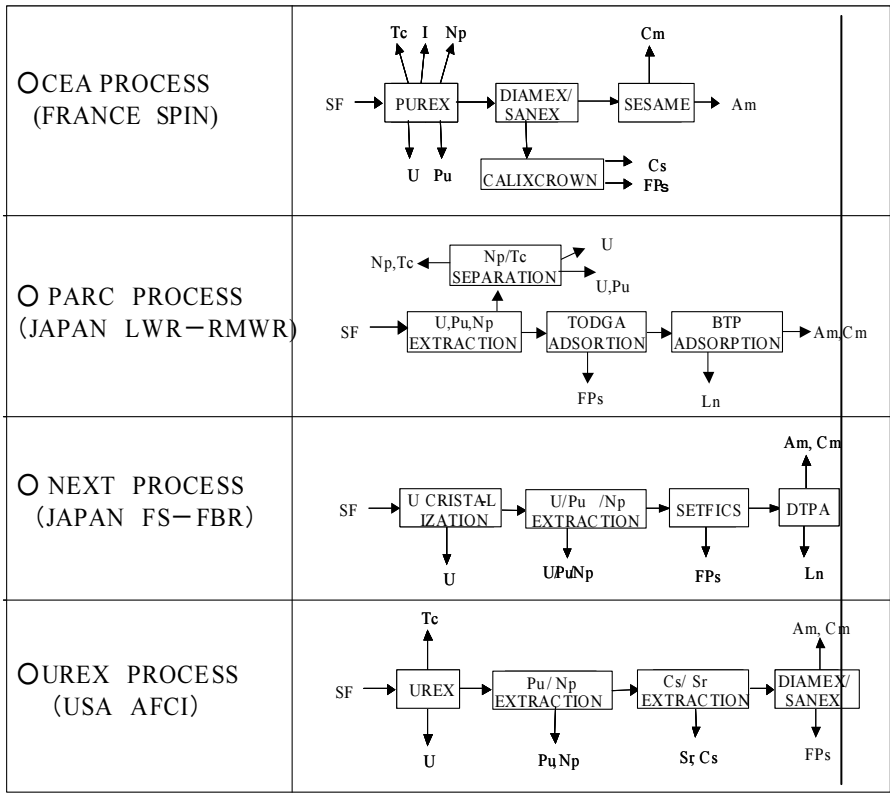

FIG. 5.6. Advanced PUREX processes with MA and LLFP separation functions.

In India, the reprocessing of SNF (U/Pu carbide) from the fast breeder test reactor (FBTR) has been successfully demonstrated in the compact reprocessing facility for advanced fuels in lead shielded cell (CORAL, Ref. [112]).

5.2.3. Russian approach of reprocessing and recycling plutonium from spent fuel and weapon material

The Russian Federation has an experience of reprocessing both reactor grade and weapon grade plutonium with producing some 1 tonne of mixed fuel (410 kg of weapon grade plutonium) (Refs [113, 114]).

The radiochemical plant RT-1 (Plant 235) of PA Mayak, was built on the site of the first military plutonium separation facility (Plant B) and brought into operation in 1976. Its first processing line was designed to reprocess HEU uranium–aluminum fuels of naval reactors. In 1978, the plant began reprocessing fuel of WWER-440 reactors. At the RT-1 plant power grade plutonium is separated as dioxide (based on Russian technology) and currently some 30 tonnes of power grade plutonium dioxide have been accumulated.

At present, the RT-1 plant processes fuels of WWER-440, BN-350/600, naval HEU fuel, and fuel of research reactors. The plant has three chopping-dissolution processing lines:

- The WWER-440 line has a name-plate capacity of 400 t_{HM}/a and a historic average throughput of 200 t_{HM}/a. But recently, the line has been operating at 25–30% of its capacity.
- The naval reactor line processes HEU uranium-aluminum fuels of naval reactors. The line's potential capacity is approximately 10 t_{HM}/a (20–30 reactor cores per year).
- The HEU-fuel line processes irradiated 90% enriched spike rods of the remaining plutonium production reactors and driver fuel of the Mayak's tritium production reactors.

In addition to the WWER-SNF, the RT-1 plant is capable of reprocessing MOX fuel of light water reactors and fuel of fast reactors producing mixed oxides based on the Granat technology (see Section 4.6.4.1).

Based on the existing projections, in the Russian Federation full scale use of plutonium is planned after 2030. By that time, it is planned to convert the excessive weapon grade plutonium and the spent fuel standard into a mixed U–Pu fuel for a closed fuel cycle.

5.2.4. French approach of plutonium recycling in PWR

It is assumed that light water reactors will dominate the production of electricity by nuclear systems during most of the current century. For this period, the development of fast reactors needs a flexible and economically acceptable plutonium management scheme.

In the following paragraphs the results of a French study [115] on Pu management are laid out shortly.

The first generation of French reactors (900 MW(e)), initially licensed to use enriched UO_2 fuel, was slightly adapted to accept plutonium fuel. For more efficient use of plutonium in PWRs several fuel concepts have been examined. The objective of these innovative fuel concepts is to facilitate core management in a Pu multi-recycling strategy and to increase fuel burnup performances, keeping safety margins the same as for current UO_2-fuelled PWRs.

The fuel concepts considered were APA[26], based on the use of inert matrices, and CORAIL[27] and MOX UE[28], using MOX. The MOX UE concept uses a homogeneous mixture of oxides

[26] Advanced Plutonium Assembly.
[27] Heterogeneous assembly using MOX and UOX fuel rods.
[28] Homogeneous assembly using MOX with LEU fuel rods.

(UO_2 and PuO_2) in each fuel rod. The CORAIL concept uses a heterogeneous arrangement of UO_2 rods and MOX rods, and the APA concept uses a heterogeneous arrangement of UO_2 rods and rods with PuO_2 in an inert matrix.

A comparative analysis of these fuel concepts was performed based on a tentative set of criteria and the impact of the introduction in the reactor park of each type of fuel.

The range of advanced assembly concepts shows that, from the reactor core physics aspect, solutions for multi-recycling of plutonium in PWRs should be possible. The available options start from a concentration of Pu in a small number of rods (APA, CORAIL), with or without recourse to an inert matrix, to total dispersion of Pu throughout the assembly (MOX EUS), with various consequences on manufacturing, Pu consumption and minor actinide production.

These fuel concepts were assessed in a simplified nuclear fleet scenario starting from the current situation up to a pseudo steady state in the future.

Assuming a nuclear park with 60 GW(e) producing 400 TW·h per year, Figures 5.7 and 5.8 show the evolution of the Pu inventory in the cycle (reactors and facilities) for the following PWR scenarios: open cycle, Pu once-through cycling, and Pu multi-recycling. In 2050, the open cycle has accumulated about 630 tonnes of Pu, and mono-recycling about 520 tonnes. For multiple recycling the Pu inventory varies between 230 tonnes (APA and MIX) and 380 tonnes according to the fuel assembly concept selected.

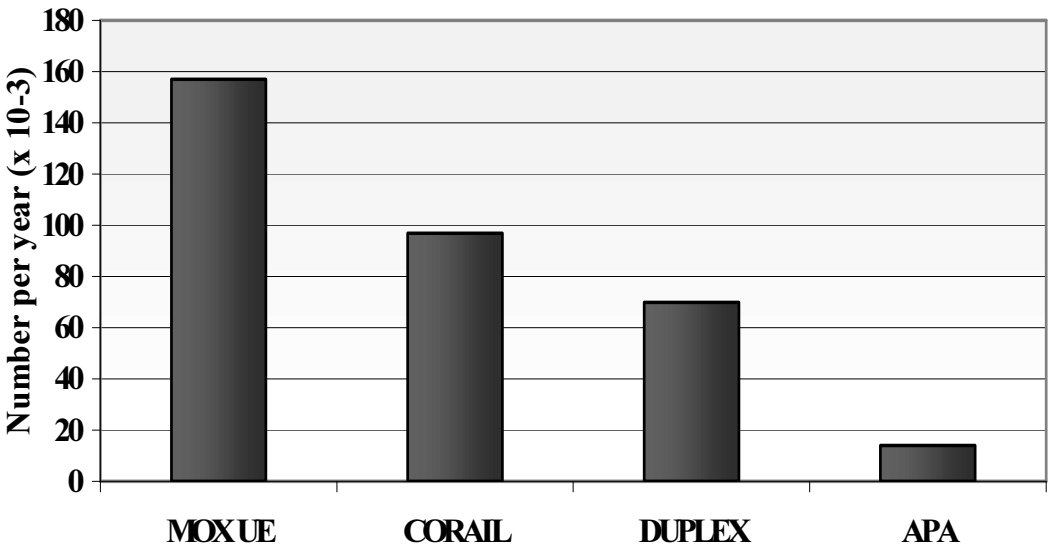

FIG. 5.7. MOX or Pu fuel rod fabrication needs.

For all this fuel concepts, studies have been performed in order to make a decision concerning their feasibility. All of these solutions require technological validation (manufacturing, behaviour under irradiation, etc.) before a decision can be made concerning their technical and economic feasibility.

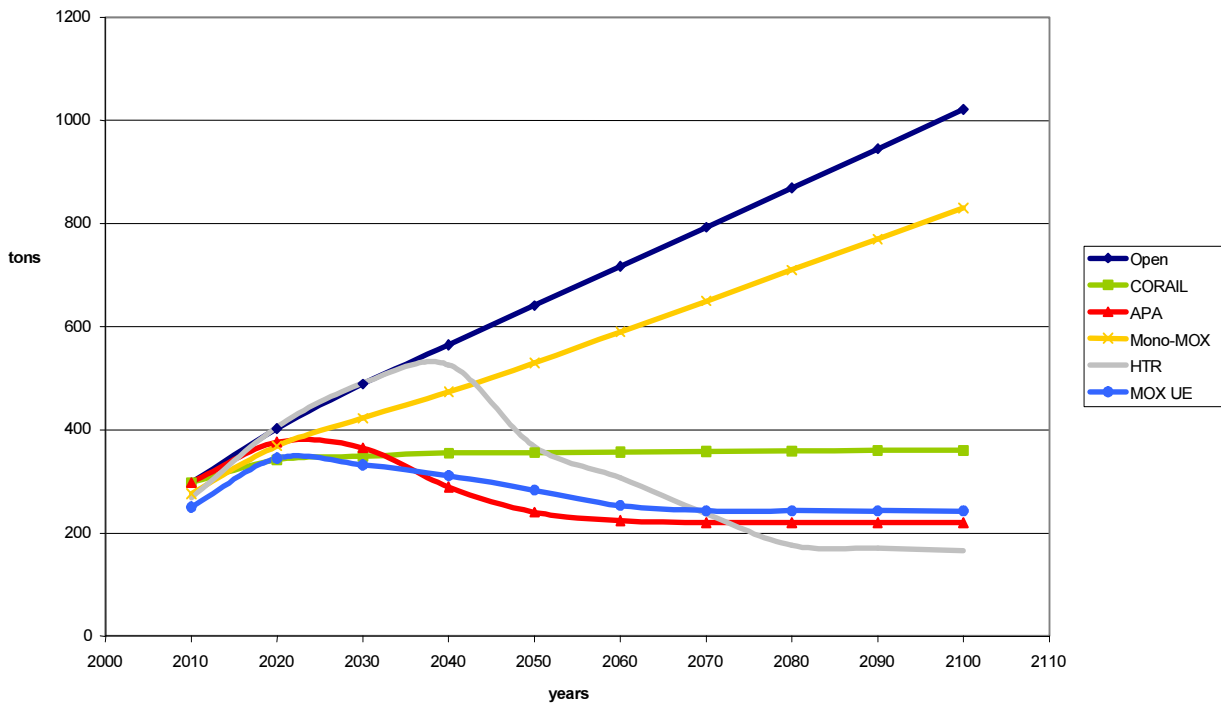

FIG. 5.8. Pu inventory (tonnes) in a nuclear park of 400 TW·h/a.

It is expected that MOX-based fuels could be ready for deployment in France by 2015 and inert matrix fuels by 2025.

Multiple recycle of Pu in PWR could enhance nuclear power by appreciably increasing the energy available from uranium resources. It could also reduce the burden of the high level waste to be disposed.

5.2.5. European approach of plutonium recycling in HTR

The technical, economic and social environment surrounding the fuel cycle may undergo profound changes in the coming decades, for example due to the increasing scarcity of uranium, significant improvements in manufacturing and reprocessing/recycling processes, and even increasingly stringent social requirements with respect to plutonium management and, in particular, waste management. Under such conditions, reactors like HTRs, which are capable of adapting to this kind of change more easily, would offer certain advantages in comparison with other reactor types.

With regard to material inventories, analyses show that an HTR core does not intrinsically burn uranium better than a PWR core (in fact, an HTR core performs only slightly better in this respect). Due to their better efficiency, however, HTRs consume less natural uranium than a PWR in an open cycle given identical electricity production (savings of 20 to 30%).

This advantage of HTRs is reflected in all material inventories. This is why, with HTRs, which exhibit conversion factors comparable to those of PWRs, the quantities of plutonium unloaded — given identical electricity production — are significantly lower (by several tens of percent) due to the gain in electrical efficiency, but also due to the high burnups which can be reached. Moreover, HTRs prove to be excellent consumers of plutonium, in particular thanks to the possibility of designing plutonium-only cores (the maximum theoretical consumption of 100 kg/TW·h_e is reached in an HTR with a plutonium-only core operating at 44% efficiency).

As has been demonstrated, the use of thorium ultimately offers little benefit with respect to natural uranium requirements, considering open cycle only (this applies to all reactor types, however). In addition, the use of thorium raises the question of the choice of fissile material to be combined with it, the possibilities being highly enriched uranium (a problem in terms of proliferation), uranium enriched to 20% (possible benefit in terms of LEU cycle) and plutonium (possible benefit in relation to Pu-only cycle). In a closed cycle, however, the use of thorium (with recycling of the ^{233}U produced) offers considerable potential for savings in natural uranium of up to a factor of 2 or 3, or even more. Moreover, the benefit of using thorium in HTRs needs to be investigated in detail with respect to the waste produced, and in particular minor actinides.

In light of the above, and in response to the problem of choosing the right fuel cycle options for HTRs, the low enriched uranium cycle clearly emerges as the most credible option for the first stage of development of HTRs.

Nevertheless, the use of HTRs as plutonium consuming cores appears an attractive option, and therefore, merits a more in-depth investigation, with R&D work on the fuel if necessary. The development of plutonium fuel should not encounter any major technological obstacles, given the results of R&D work already carried out on such fuel in the past. In the longer term (over the next few decades), assuming the likely hypothesis of progressive exhaustion of exploitable reserves of natural uranium, thorium could become an attractive resource in closed cycles. In addition, as has been seen, thorium-based cycles may prove beneficial in terms of the associated radioactive waste (significantly smaller quantities of minor actinides than with uranium–plutonium cycles). It would therefore be useful to conduct preliminary studies on this issue, bearing in mind nevertheless that, in technological terms, the industrial development of such cycles would necessitate considerable R&D effort (in relation to manufacturing technique and reprocessing/recycling processes).

The fact remains, however, that the results presented above are not only old but also incomplete, due to loss of know-how over time and difficulties in locating all related documentation. In addition, the results relate only to large power reactors (around 1000 MW(e) or more), whereas current considerations for HTR encompass only small-size modular reactors (100–300 MW(e)), which may additionally exhibit different core geometries (for example the annular core of the GT-MHR). This should not alter the principal trends identified here, but could significantly affect certain figures used as a basis for this analysis.

In summary, it is clear that a considerable amount of work still needs to be done toward a detailed study of the different fuel cycles for HTRs, even if only the cycles deemed to be potentially beneficial are considered.

Theoretical Pu consumption in reactors

Assuming a fission energy of 200 MeV (1 ev = 1.6 10^{-19}J), it can easily be calculated that complete fission of 1 kg of ^{239}Pu releases the following quantity of energy: 2.24 10^{-2} TW·h thermal, equivalent to approximately 44 kg/TW·h thermal.

The possible maximum theoretical plutonium consumption levels are given in Table 5.4 as a function of the electrical efficiency of the reactors considered:

TABLE 5.4. PLUTONIUM CONSUMPTION LEVEL

Electrical efficiency	Reactor type	Maximum theoretical plutonium consumption in kg/TW·h_e
33%	PWR	133
40%	FBR	110
47%	HTR	94

5.2.6. *Experimental reprocessing technologies*

Non-aqueous processing methods

Since the early days of the nuclear industry, various pyrochemical processes have been investigated to separate actinides from SNF. Pyrochemical separations often rely on electro-refining or electro-winning techniques, in which fuel is preferentially separated or electrolytically dissolved from cladding into molten salts, and then recovered on some type of electrode.

Studies and tests on pyrochemical recycle techniques are currently performed throughout the world; mainly in France, Japan, the Republic of Korea, the Russian Federation (RIAR) and the USA (ANL, [141]). In the past, in the USA the focus of this work was on the application of pyrochemical processing to recycle fast reactor metal fuel (e.g. around 3 tonnes of EBR-II driver and blanket fuel has been reprocessed at INL, USA) and to incorporate spent LWR fuel as feed material into the fast reactor fuel cycle. Programmes in the Russian Federation have applied pyrochemical techniques to the preparation and recycle of oxide fuel for fast reactor application. Programmes in Japan are broader, addressing recycle of metal, nitride, and oxide fuels and the incorporation of spent LWR fuel as feed material into the fuel cycles.

Alternative non-aqueous processes currently under consideration to be applied in SNF processing include volatility and reductive extraction processes, such as chloride volatility and fluoride volatility. Such processes can be helpful in the processing of complex fuel types, including inert matrix fuels. Reductive extraction processes investigate replacement reactions to separate certain fission products and actinides. Molten metal/molten salt systems are used for the application of reductive extraction processes.

One example is the Japanese fluoride volatility process (FLUOREX, Fig. 5.9) that was developed to reprocess PWR spent fuel efficiently [116]. In this process, most of uranium from SNF is removed first before purification of the mixture of plutonium and residual uranium by solvent extraction from the fission products. This process is being developed to recycle metal fuel of future fast reactors [117].

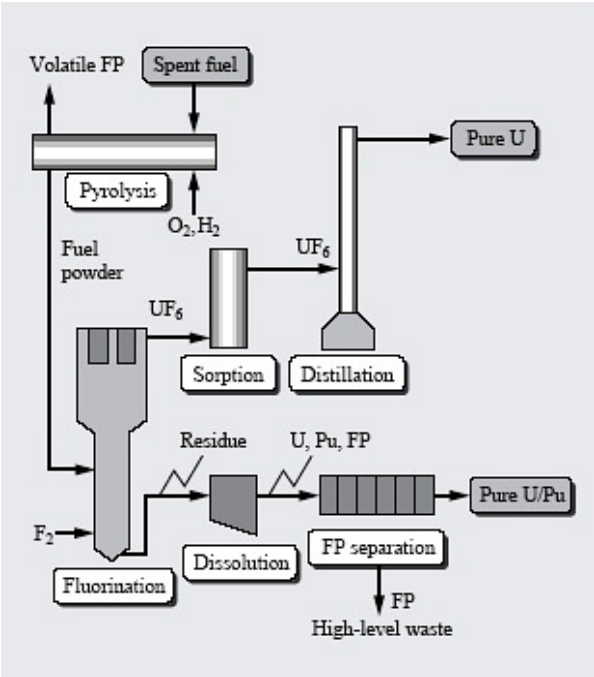

FIG. 5.9. Typical flow of FLUOREX reprocessing in Japan.

In France, experiments have shown that fluoride melts are superior to chloride melts for selective separation of An/Ln.

During the 1990s, new concepts for the transmutation of fission products and the associated fuel cycles came up. Most of the variants considered require multiple recycling of targets or fuel to obtain high transmutation rates and thus a sufficient reduction in the radiotoxicity. The far future options, such as dedicated fuel cycles and transmutation in molten salt reactors provide further perspectives for the exploration of these processes. This is obvious in the case of molten salt reactors, for which pyrochemical techniques are the natural, and perhaps the only, possible reprocessing solution.

Processes for coated particle fuels

Already some decades ago coated particle fuel reprocessing was investigated in Germany and the United States of America because this fuel type was used in the HTRs of that time. This fuel contained highly enriched uranium (HEU) and thorium as mixed oxides/carbides or with separate fuel and breeding particles embedded in the fuel matrix. The extracted uranium contains mainly ^{233}U, with other radioactive U isotopes requiring remote fabrication of new fuel from this resource.

The main steps of HTR fuel reprocessing are:

- head-end separation of graphite and fuel kernel;
- chemical separation of the different mass streams;
- back end purification of products;
- re-conversion of regenerated fissile materials.

The development programme in Germany included the operation of a pilot plant that processed more than 10000 unirradiated fuel pebbles of different types (HEU-BISO, HEU-TRISO, LEU-TRISO, graphite balls) showing a very reliable operation and effective performance. However, work with irradiated materials was not started because the back end strategy for the HTR was changed to direct disposal instead of reprocessing. As already mentioned in the Section 2.4.1, the whole HTR development in Germany was stopped in 1988.

In Japan, JAEA started construction of the HTTR (high temperature engineering test reactor), which is a high temperature gas cooled reactor employing the TRISO coated-particle type fuel system, and its first criticality was attained in November 1998. JAEA has considered several back end options for the HTR fuel, including the possibility of reprocessing. Reprocessing process for the HTR was studied during the late 1970s and early 1980s. Some head-end reprocessing techniques were investigated for applying the conventional PUREX process to the HTR fuel. JAEA demonstrated the feasibility of the graphited-CO_2 reaction and jet grind methods as head-end reprocessing techniques by bench scale experiments.

5.2.7. Future reprocessing perspectives

Aqueous processes have a high potential for the future to reprocess SNF from commercial LWRs, including the separation of minor actinides (MAs). This was demonstrated by extensive research, especially in Europe and Japan. The major technical drawbacks of this technique are:

- the limited solubility of advanced fuel forms;
- the limited stability of the organic extraction molecules in high radiation fields.

As already mentioned above, there is also a reasonable potential for the application of aqueous processes (THOREX-type) for future commercial use in new HTRs and also in other advanced systems like the advanced heavy water reactor (AHWR) considered in the three stages Indian nuclear power programme (Fig. 2.13 in Section 2.3.5) [18].

Pyrochemical techniques, up to now developed only at a laboratory scale or pilot scale, show an interesting potential for reprocessing of advanced fuels due to their good compatibility with most fuel forms and their high radiation resistance. In addition, the increased proliferation resistance and the compactness of the technique are important advantages. However, the feasibility of the recovery of minor actinides still needs to be investigated. Electro-refining is considered to be the most promising pyrochemical method, and it is being investigated worldwide, especially in the USA, Japan, the Republic of Korea (Fig. 5.10), and the Russian Federation.

Aqueous and pyrochemical recycle techniques should be considered as complementary technologies. For example aqueous processes might be used to recycle the first LWR cycle, including the MA. In a second step, the fast breeder or burner reactor could rely on dry recycle techniques.

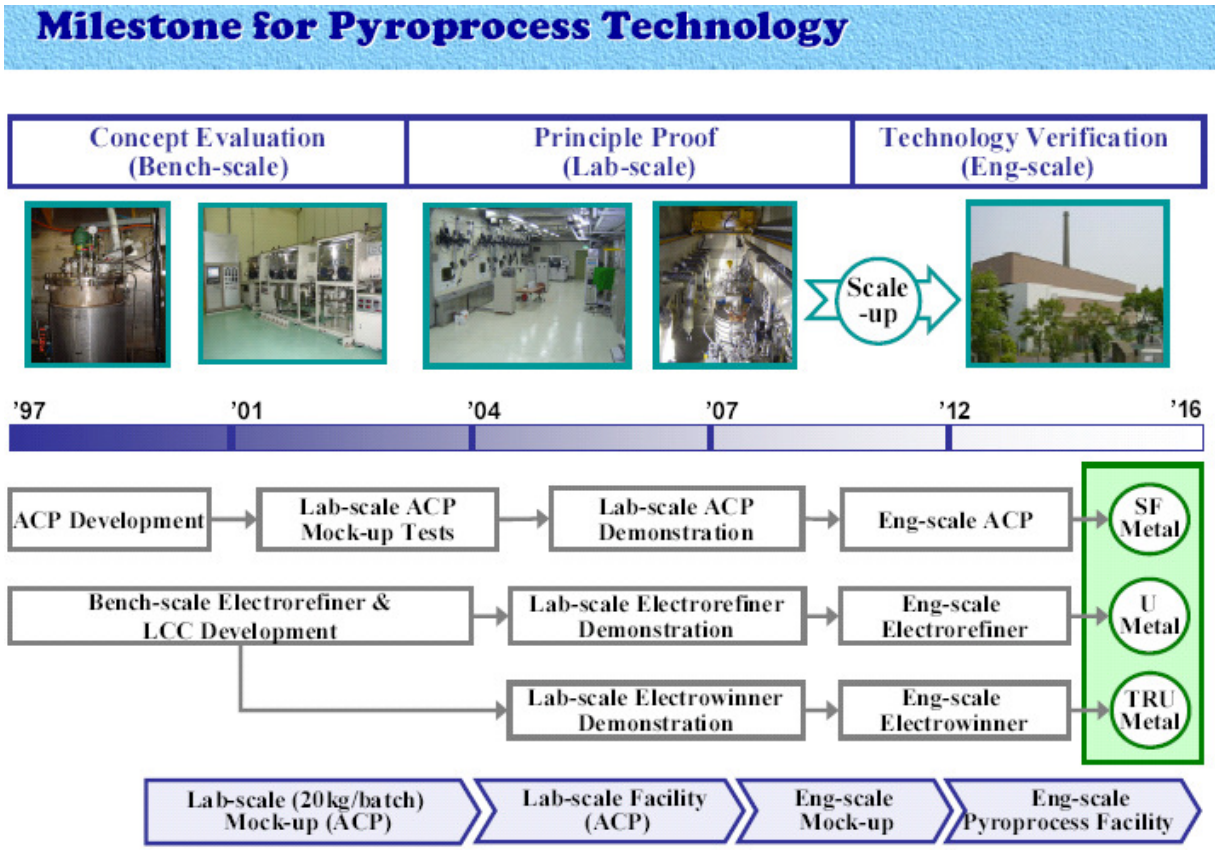

FIG. 5.10. Korean development of pyroprocessing technology (IAEA TWGNFCO 2006) (ACP = Advanced conditioning process).

Recycling of reprocessed uranium (RepU) has been used in the past and is currently used in several countries (Belgium, France, Germany, India, Japan, the Netherlands, the Russian Federation, Sweden, Switzerland, the UK). However, RepU contributes only about 2000 tonnes U to a total market of 65000 t·U per annum; it could become an interesting technology in the future depending primarily on the price of natural uranium [50]. It would reduce the radiotoxicity and the volume of waste, and increase available fertile and fissile

resources; but it has also some disadvantages, e.g. limitation of burnup, and increased decay heat.

5.2.8. Transportation of nuclear material during recycling

Plutonium and MOX fuel have been safely transported for about 40 years, mainly by trains or trucks, but also by ships and airplanes.

Shipments of plutonium and MOX have to fulfil the highest physical safety and security requirements. All packages and transport operations must comply with the applicable national and international transport safety laws and regulations which are practically all based on the transport safety regulations recommended by the IAEA.

Physical protection requirements are provided in the IAEA document [118] and are further detailed in national regulations and guidelines which are classified 'restricted'.

For MOX shipments a large number of technical and administrative requirements have to be fulfilled. In addition, for maritime transports, the INF Code (Code for the Safe Carriage of Irradiated Nuclear Fuel, Plutonium and High Level Wastes in Flasks on board of Ships) is typically applied to all transports covered by this voluntary code. The International Maritime Organization (IMO) is currently taking the necessary steps to make the IAEA requirements [118] mandatory under the provisions of the Safety of Life at Sea Convention.

PuO_2 powder is filled in cans containing about 3 kg plutonium dioxide. According to an IAEA document [119], PuO_2 and MOX fuel must be packaged in accident resistant Type B containers, which also take into consideration the fissile characteristics of the contents. BNFL, COGEMA and others have developed packages for PuO_2 powder transport. The BNFL 1680 and the COGEMA FS47 packages permit the shipment of large quantities of plutonium oxide in powder form. Similarly, packages designed for one or more MOX fuel assemblies have been produced and are currently in use.

Plutonium transport packaging calls for diverse packaging types suitable for its many forms, ranging from powder to complete MOX fuel assemblies for LWRs and FRs. Each type of package and design must satisfy the complete set of regulatory design and performance requirements. Although the material may be in very different forms the containment performance requirements for the packaging are the same. All packages for PuO_2 and MOX transportation which contain Category I quantities of plutonium must be designed to meet the Type B criteria. That means they are safe under both normal and accident conditions of transport. Additionally, their designs must be approved to account for the fissile nature of the contents.

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

Additionally, for road transport, special security trucks are typically used. Irradiated MOX fuel moved from NPPs to reprocessing facilities is transported in Type B(U)F packages which are heavily shielded and which may be co-loaded with uranium fuel due to residual heat and neutron radiation considerations.

It can be stated that today the transportation of all kinds of fuel material used during recycling is technically and legally proven and mature. There have never been accidents of concern in this field and there is no need for special developments for the near future. Nevertheless, a continuous adjustment to the general development of transportation technologies and the relevant national and international laws has to be performed. With the development of new

types of fuel in particular the packaging technologies have to be revised to fulfil the specific requirements for those new fuel types.

5.3. Final disposal of nuclear waste

5.3.1. General aspects

There is no step of the whole NFC that is more under the pressure caused by politics and public debates than the final disposal of high level waste (SNF and process waste from reprocessing).

Regardless which options of back end fuel cycle have been chosen — once-through or reprocessing/recycling — there remains a challenge in the final disposal of SNF and/or radioactive waste. Some countries are working since decades on respective national repository programmes to dispose of the SNF and radioactive wastes in geologic repositories. Here, the real challenge is how to engage the public and stakeholders for their acceptance of the high level wastes management programme including a geologic repository, on a national basis, as well as on a cooperative regional or international arrangement.

Some years ago the IAEA released a document [120] providing a generic approach to radioactive waste management, identifying potential final disposal options for various waste categories. Thus, geological disposal is required for the most active high level waste (HLW), spent nuclear fuel (SNF) and long lived low and intermediate radioactive waste (LILW), with radionuclides that may take tens of thousands to hundreds of thousands of years to decay to acceptably low levels. Near surface disposal is an option suitable for short lived LILW that contains mainly radionuclides, which decay to radiologically insignificant levels within a few decades or centuries. LILW that contains limited concentrations of long lived radionuclides may also be suitable for near surface disposal.

A survey on the national and international situation with respect to final disposal of nuclear waste recently was provided on an international conference, organized by the IAEA in cooperation with the EC and the OECD/NEA [121]. This review focused on HLW and long lived LILW that needs far from surface disposal; another review discusses LILW that can be disposed in near surface facilities [122].

At the above-mentioned conference on the one side the progress was noted to establish geological repositories for SNF and HLW, in particular in Finland and in the USA. This also raised the issue that time has come for the IAEA to clarify the requirements for safeguarding of such repositories.

On the other side, it was realized that long term (interim) surface storage is becoming a reality in many countries, due to various reasons, including delays in final repository programmes, lack of resources, lack of public or political acceptance and finally also due to uncertainties if SNF should be considered as waste or as a future resource. It was clearly stated that, in the long term, surface storage is not a sustainable solution. Finally, the problem of bringing disused sealed sources under regulatory control was considered [123].

Several countries reprocess their SNF (Section 5.2). As a result, they produce and hold separated plutonium. Part of this separated (civil) plutonium is currently recycled as MOX fuel in light water reactors, but another part is stored pending a decision on disposition. That brings up the issue of proliferation protection.

There is an increasing interest among several countries in strengthening research efforts and development (R&D) programmes on advanced and innovative fuel cycles to deal with these fuel cycle challenges. The new concepts should introduce new features and special attributes

to enhance proliferation resistance, resource utilization, environmental protection, safety, and wastes' reduction. However, it might be still a long way to a technological maturity and readiness of these concepts.

Within the European Commission project Red Impact the consequences of P&T on geological disposal have been investigated since 2004 [124].

Final disposal will also be necessary in case that separation and transmutation would reduce the waste volumes.

5.3.2. Overview on national concepts for long term disposition of HLW

Presently, every country with a nuclear power programme has a radioactive waste management programme. In all of these countries geological disposal is considered to be the preferred option. Many of these countries also have laws regulating the export, and, more importantly, the import of radioactive waste, which enforces a national solution for the final disposal.

Disposal in space is a high risk alternative and disposal under the seabed has been prohibited by international conventions.

The international association for Environmentally Safe Disposal of Radioactive Material (EDRAM) has been established in order to follow a common strategy for ensuring the long term safety of a system for disposing of high level waste in all national radioactive waste management programmes. In the EDRAM member states special companies/institutions take care of the development and deployment of the technical solutions for final disposals: ENRESA (Spain), ANDRA (France), ONDRAF (Belgium), NAGRA (Switzerland), Ontario Power (Canada), OCRWM (USA), POSIVA (Finland), NDA (UK), DBE and BfS (Germany), NUMO (Japan) and SKB (Sweden).

There are underground rock laboratories (URLs) in the following EDRAM member countries: Belgium, Canada, Finland, France, Germany, Japan, Sweden, Switzerland and the USA. Research is underway there on different geological formation, e.g. clay, salt and crystalline rocks.

In Table 5.5 an overview is given on some selected IAEA Member States' disposal concepts for high level radioactive wastes including direct disposal of SNF.

TABLE 5.5. MEMBER STATES' DISPOSAL CONCEPTS FOR HIGH LEVEL RADIOACTIVE WASTES INCLUDING SNF

Member States	Belgium	Canada	Czech Republic	Finland	France
Waste form	HLW, VHLW	SNF	SNF	SNF	VHLW
Over pack - Dimension(m) - Capacity - Material - Life time(as)	- 0.50ϕx2.68 - 2 canisters - Carbon steel - >2000	- 1.2ϕx3.9 (one of several designs) - 324 used CANDU fuel bundles - Cu corrosion barrier with steel insert - 100,000	- EDU (440) 3.237 ϕx 0.368; ETE(1000) 4.720 ϕx 0.423 - 7FA (EDU) or 3FA (ETE) - Carbon steel - 5000	- 1.05ϕx4.8(BWR) - 12 BWR FA or 4 PWR (EPR) or 12 PWR (WWER-440) - Copper - 100000	- 1.607 ϕx 0.590 - 1 primary waste package - Carbon steel - 1 000–4 000
Repository - Capacity - Host rock - Depth - Emplacement	- 1150m^3 HLW ; 625m^3 - Boom clay - 230 m - Horizontal	- 3.6 million CANDU bundles (design) - Crystalline rock or sedimentary rock - 500m (design will depend on-site conditions) - Vertical in-floor or horizontal tunnel	- 3600t - Granite - 500m - Horizontal/ vertical	- Olkiluoto-1/2, 2530 t·U Olkiluoto-3, 1980 t·U Loviisa1-2 , 1020 t·U - Crystalline rock - 420 m - Vertical	- 6300m^3 - Clay (argillite) - 500m - Microtunnels
Current status	R&D	R&D	SS	SC	Design and preparation of application to be provided in 2014
Operation	2040/2060 for B-waste 2090/2100 for C-waste	2035 (earliest estimated in-service date for financial planning purposes)	2065	2020	2025

as – years; HLW – High level waste from reprocessing; CS – Carbon steel; SNF – spent nuclear fuel; SR – Soft rock;
HR – Hard rock; SR – Sedimentary rock; HR – Hard rock; T or C – Tunnel or cavern disposal;
GS – Geological study; SS – Site screening; SC – Site characterization; VHLW – Vitrified high level waste;
LA – Licensing application for construction permit; can. – Canisters; EDU = NPP Dukovany NPP, ETE = NPP Temelín; FA – Fuel assemblies.

TABLE 5.5. MEMBER STATES' DISPOSAL CONCEPTS FOR HIGH LEVEL RADIOACTIVE WASTES INCLUDING SNF (cont.)

Member States	Japan	Republic of Korea	Russian Federation	Sweden	Switzerland	USA
Waste form	HLW	SNF (PWR, CANDU)	HLW	SNF	SNF (UO2 and MOX)/HLW	SNF, HLW
Over pack -Dimension(m) - Capacity - Material - Life time(as)	 - 0.82ϕx1.73 (ref. option) - HLW corresponding to 8.6 x 10^{-2} m^3/ FA(PWR) - Carbon steel (reference option) - 1000 (ref. option)	 - 1.02ϕx4.83 - 4 PWR FA./ 297 CANDU bundles - Cast iron insert + Cu outer shell - 1000		 - 5 x 1.75 - 1-2 BWR or 4 PWR FA - Copper - 100000	- Cylinder, diameter/length: 0.94/3.25 Volume: 2.26 m^3 for two 180-l-HLW canisters Diameter/length: 1.05 /4.92, Volume: 4.26 m^3 for max. 4 PWR or 9 BWR FA's - Capacity see above - Carbon steel -10000 years	 - 1.644ϕx5.16 - 21PWR FA - Alloy 22 - 10000
Repository - Capacity - Host rock - Depth - Emplacement	 - More than 40 000 canisters - SR/HR - More than 300m - Vertical./Horizontal	- 36 000 t (PWR 20000 t + CANDU 16000 t) - Cristalline rock - 500m - Vertical	 - HR - 100 to 1000m	 - 9000 t - crystalline rock (granite) - 500 m - Vertical/ Horizontal	- SNF from 2435 t·U , HLW from 1140 t·U, volume of packaged waste (SNF and HLW): 7325 m^3. - Clay rich sedimentary rock (OPA) - 500-900m - Horizontal	- 63 000 t - Tuff - 300 m - Drift
Current status	Siting	R&D	R&D	SC	Siting	LA
Operation	For 50 years from around 2035		2030	2020		2020

HLW - High level waste from reprocessing; CS – Carbon steel; SNF – Spent nuclear fuel; SR – Soft rock;
HR – Hard rock; SR – Sedimentary rock; HR – Hard rock; T or C – Tunnel or cavern disposal;
GS – Geological study; SS – Site screening; SC – Site characterization; VHLW – Vitrified high level waste;
LA – Licensing application for construction permit; can. – Canisters; FA – Fuel assemblies.

5.3.3. Near surface final disposal of low and intermediate radioactive waste (LILW)

As already mentioned in Section 5.3.1 near surface disposal of LILW is considered to be an option under certain conditions [122]. There are two main conditions that are to be considered to make a near surface disposal acceptable:

- The waste mainly contains radionuclides which decay to radiological insignificant levels within a defined limited time (some decades up to some hundred years).
- The disposal concept provides a convincing barrier system.

The basic safety philosophy for the near surface disposal of radioactive waste is to isolate the waste from the environment for a sufficiently long time to allow the decay of the shorter lived radionuclides and, in the longer term, to limit releases of residual radionuclides into the environment by relying on multiple barriers.

Extensive experience that has been collected from operation of near surface facilities, which provides a good basis to understand the evolution of disposal systems and the behaviour of radionuclides in respect of their mobilization and subsequent migration through both engineered and natural barriers. This understanding has been used to develop models for the performance of near to surface disposal systems.

For close to the surface repositories (within a few meters) institutional controls are needed to assure the adequate performance of the waste isolation barriers during the initial period when the activity of short lived radionuclides is still high. The anticipated duration of institutional controls is an important strategic decision with significant implications for the development of the disposal system, including the definition of waste acceptance criteria. If disposal is located at greater depths (tens of meters), i.e. in a rock cavity or moderately deep borehole, less reliance may be placed on institutional controls. Near surface disposal of LILW has been practiced for many decades and there are more than 80 near surface repositories around the world.

It can be expected that many new near surface facilities will be constructed and many existing facilities will be upgraded in the next few decades, especially in developing countries, eastern European countries and the countries of the former USSR.

5.3.4. Waste transportation

Legislatives for the transportation of radioactive materials in the considered countries (CIS and Eastern Europe) are based on the IAEA regulations. Transport systems for low and intermediate level wastes have not been developed on a larger scale. This was not necessary because the management and the storage took place at the NPP sites. In most cases only the radioactive waste from small waste producers has to be transported to the centralized management and storage facilities. These shipments are and were done most on road by truck. For the transportation diverse packages are used according to IAEA regulations for industry packages IP 2, Type-A and Type-B containers and others.

In view of the future installation of centralized storage and repository sites also the transportation systems will have to be modernized and qualified.

In connection with the management and disposal of wastes from naval bases of the Russian Northern Fleet also packaging and transportation tasks must be solved. An example is the conditioning of wastes in drums and the adjacent shipment and storage of these drums within cylindrical steel containers.

5.4. Partitioning and transmutation (P&T)

5.4.1. General aspects

The basic idea behind partitioning & transmutation (P&T) is to relieve the waste management of the big burden of the transuranium actinides with very high lifetimes (in particular the so-called minor actinides: neptunium, americium and curium), formed by neutron capture in today's reactors [126].

Most radionuclides formed in the fuel by fission decay relatively rapidly, so that their collective radioactivity is reduced to less than 0.1% of the original level 50 years after being removed from the reactor. In contrast, some of the actinide isotopes contribute to the radioactivity of the waste for thousands of years and more. P&T technologies aim at first separating (partition) those actinides with high lifetimes by special processes from other short life isotopes during reprocessing and then destruct (transmute) the actinides (and other longer lived fission products such as ^{99}Tc and ^{129}I) by fission or neutron capture, ending up with short lived or even stable isotopes.

This transmutation can be achieved in conventional reactors[29] or in Accelerator Driven Systems[30] (ADS, see also Section 2.7.1), where in principle highly accelerated nuclear particles like protons are bombarding a target to generate high energetic neutrons (by spallation reactions) by which for example actinides are transformed into less heavy isotopes with much shorter half life time.

Based on the performed studies so far it is estimated that transmutation can reduce the time needed for a geological (final) disposal of HLW to reach radiation levels of natural uranium ore by at least 3 orders of magnitude. However, it is essential to note that the elimination of the minor actinides does not yield a radiological benefit unless the plutonium is also eliminated.

Recent studies in Japan and the Republic of Korea have shown that P&T technology also has the potential to reduce the necessary space for storing HLW in geological repositories (Refs [127, 128]).

Obviously, long term scenarios are to be considered in which thermal and fast reactors are both used for the recycling and incineration of the actinides.

5.4.2. Technological concepts of ADS for P&T

These systems consist of three main units — the accelerator, target/blanket and separation units (Fig. 5.11 taken from Ref. [129]). The accelerator generates high energy (around 1 GeV) charged particles (e.g. protons), which hit a heavy material target. This bombardment leads to the production of a very intense neutron source (a process called spallation). These neutrons enter a subcritical core (often called a blanket) where they can be multiplied. In the core, the transmutation of actinides and fission products takes place. After a time, already transmuted nuclei have to be removed from the fuel in order to avoid their undesirable activation. Long lived fission products and actinides return from the separation unit to the blanket. Short lived and stable isotopes as well as fission poisons are removed and processed for storage.

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[29] In France, an intensive irradiation programme of MA bearing inert matrix targets has been performed in fast reactors [139].

[30] In Reference [140] a comparison of ADS and fast reactors for P&T is presented.

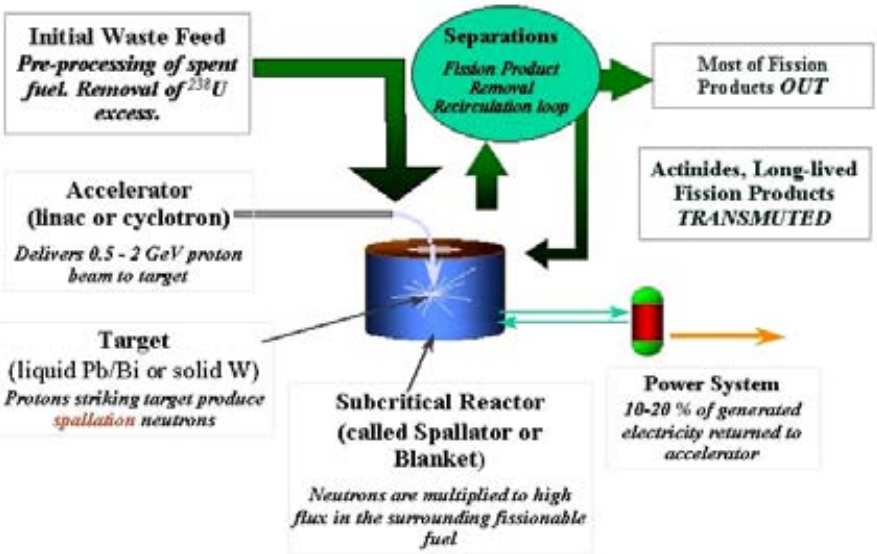

FIG. 5.11. Partitioning and transmutation flow sheet.

The idea to use accelerator driven systems is not new. But early developments, e.g. at the Lawrence Livermore Radiation Laboratory in the 1940s were cancelled in 1952. At that time ADS was developed to produce fissile materials, namely ^{239}Pu from ^{238}U or ^{233}U from ^{232}Th. But then high grade uranium ores were discovered in the United States of America. In addition, the idea of exploiting the spallation process to transmute actinides and fission products directly turned out to be ineffective with the technology available at that time.

Later, new developments in accelerator technology have started a new wave of interest in using ADS. In the last decade, hybrid accelerator-driven systems have been proposed for different purposes. Several US national laboratories like Los Alamos National Lab, Oak Ridge National Lab, and ND Pacific Northwest Lab (Fig. 5.12) have developed several ideas to use hybrid system with a linear accelerator based on thermal and recently also on fast neutrons for incineration of plutonium and higher actinides, for transmutation of some fission products as well as for gaining energy.

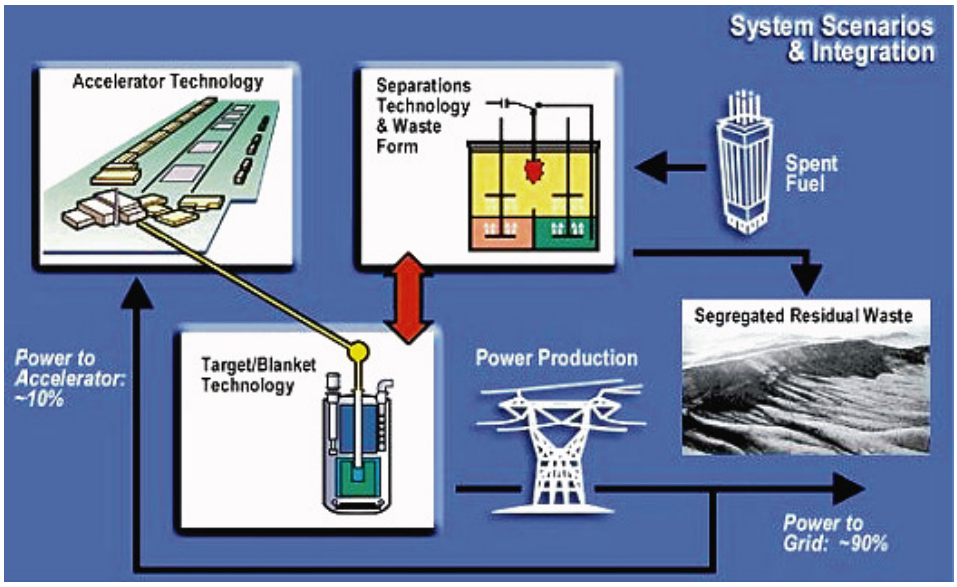

FIG. 5.12. Accelerator transmutation of waste (ATW).

Some years ago a CERN team lead by Carlo Rubbia (Fig. 5.13) proposed a cyclotron-based hybrid system to produce nuclear energy based on a thorium fuel cycle [130].

This concept is supposed to be an attractive option reducing the concerns about higher actinides in the SNF and offering the possibility of utilising cheap and quite abundant thorium.

Accelerator-driven systems have unique characteristics to meet the requirements of safe operation and non-proliferation. For example, ADS resolves the safety problems associated with criticality, because ADS operates in a non-self-sustained chain reaction mode and never achieves a self-sustained reaction. They are therefore also called undercritical systems. Also, low inventory of radioactive elements in reacting volume and low pressure and moderate system temperatures are important positive safety features of ADS systems. In principle, ADS can work without additional safe shutdown mechanisms (control rods) and accept fuels that would not be acceptable in current critical systems. At least potentially those systems may reduce public concerns on nuclear energy technologies.

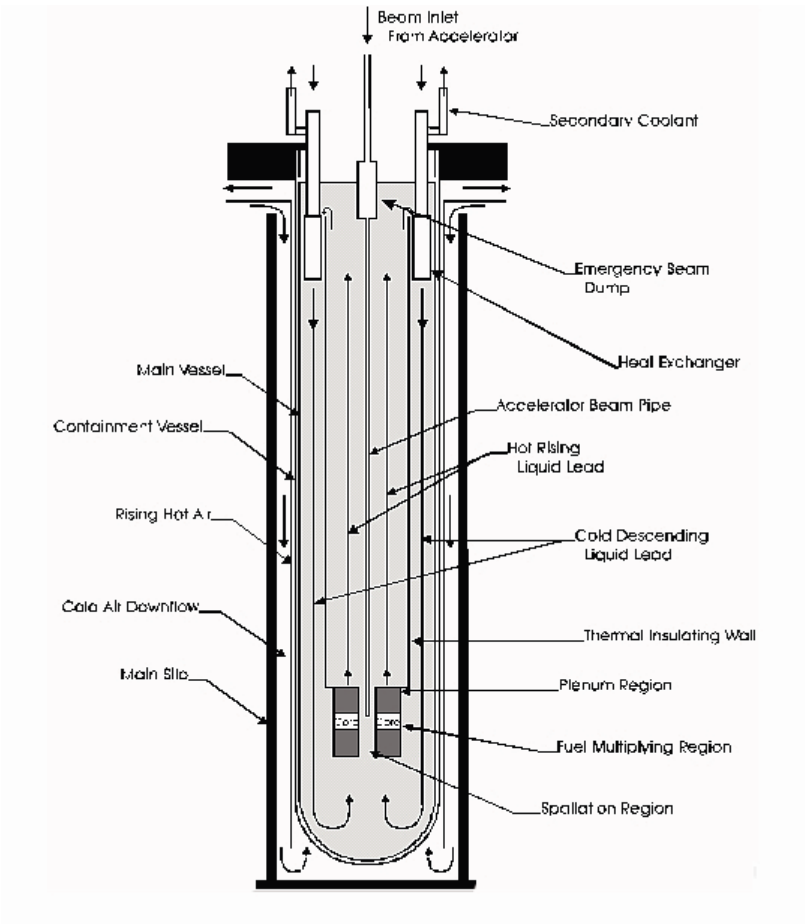

FIG. 5.13. Target and multiplying assembly from Rubbia.

In the meanwhile, many investigations on ADS systems are underway worldwide, in particular in Europe (Belgium and France), the Russian Federation and the USA, and in the meanwhile also in the Far East (China, India, Japan and the Republic of Korea).

In order to get more transparency and potential for international cooperations into these activities, the IAEA took an initiative to bring those activities together. An Advisory Group Meeting was held in 1999 in Taejon, the Republic of Korea, to review jointly the national accelerator driven system programmes in the area of partitioning and transmutation [131].

The results of this review are laid out shortly in the following sections for Europe, Asia, the Russian Federation and the USA.

5.4.2.1. ADS/P&T research in Europe

The main driving force is long lived waste transmutation, but the ADS capability to produce energy is also investigated. Driven by the establishment of the European Industrial Partnership (EIP) to advance the engineering design studies of a ~100 MW(th) ADS demonstration facility, the national programmes on ADS R&D are converging towards the demonstration of the basic aspects of the ADS concept. These R&D activities are conducted both nationally and as joint efforts under the umbrella of the EU. Presently, the EIP consists of leading European nuclear industrial companies from six countries.

The fuel cycle strategy and the fuel cycle back end closure is a national policy in Europe.

There are various strategies considered:

- UO_2 in LWR + final disposal;
- UO_2 in LWR + ADS + final disposal;
- UO_2 in LWR + MOX in LWR + ADS + final disposal;
- UO_2 in LWR + MOX in LWR + (dirty MOX + MAs) in FbR + ADS + final disposal.

In nearly all scenarios, P&T and ADS are given consideration that leads to a need for coordinated R&D.

Both critical and subcritical reactors are potential candidates as dedicated transmutation systems. Nevertheless, critical reactors heavily loaded with fuel containing large amount of minor actinides (Am and Cm) pose safety problems caused by unfavourable reactivity coefficients and small delayed neutron fraction. With regard to this latter problem, the subcriticality is particularly favourable and allows a maximum load of minor actinides per unit while operating in a safe manner.

The European Technical Working Group (ETWG) on ADS under the chairmanship of Prof. Carlo Rubbia played a coordinating role at European level for P&T and ADS development as a route for waste management. The ETWG members concluded in their report of April 2001 that P&T associated to ADS could help the waste management problem; there is a need for a first step demonstration of ADS at international level and there is a need for a coordinated R&D effort at European level with a strong support from the EC.

5.4.2.2. ADS/P&T research in Asia

ADS R&D studies are pursuing both goals: energy production with reduced waste production and decreased proliferation hazard, on the one hand, and long lived waste transmutation, on the other hand. The programmes are presently conducted at national level, with some bilateral or multilateral cooperation agreements.

Research in Japan

JAEA has started a basic R&D programme to validate the engineering feasibility of the ADS development since 2002 (Figs 5.14 and 5.15, Ref. [132]).

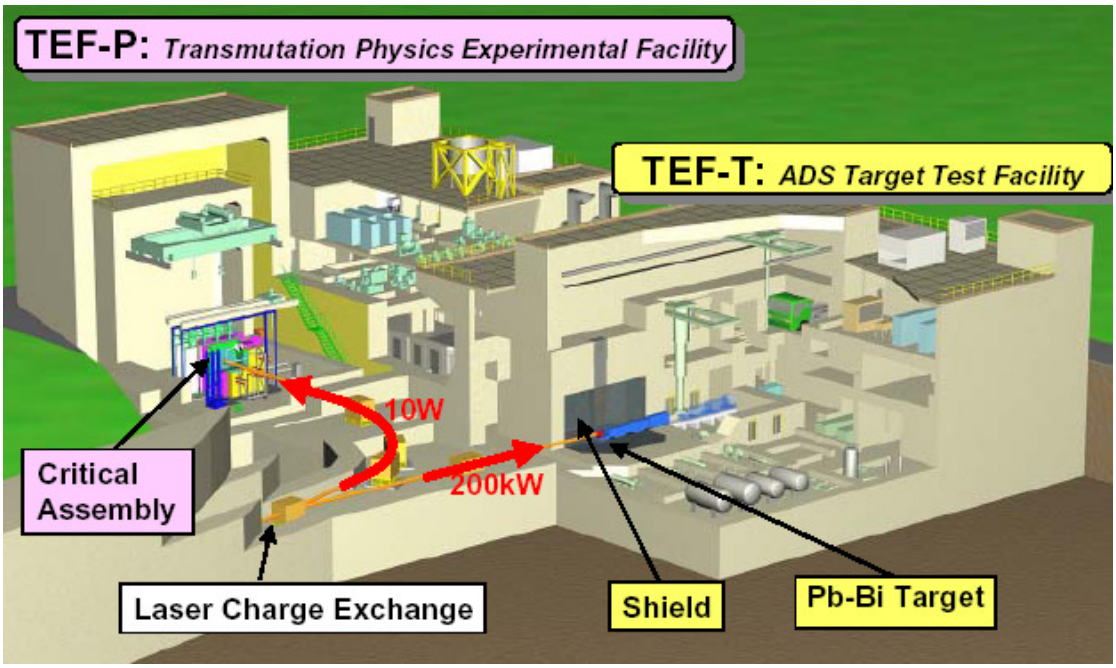

FIG. 5.14. Transmutation experimental facilities in Japan.

After the basic R&D for subcritical core design and technology, lead–bismuth eutectic technology and super conducting linear accelerator, an experimental ADS with 80 MW thermal power is planned in the late 2010s to demonstrate the engineering feasibility of the ADS. The experimental ADS will be operated by MOX fuel at first and gradually altered to MA nitride fuel.

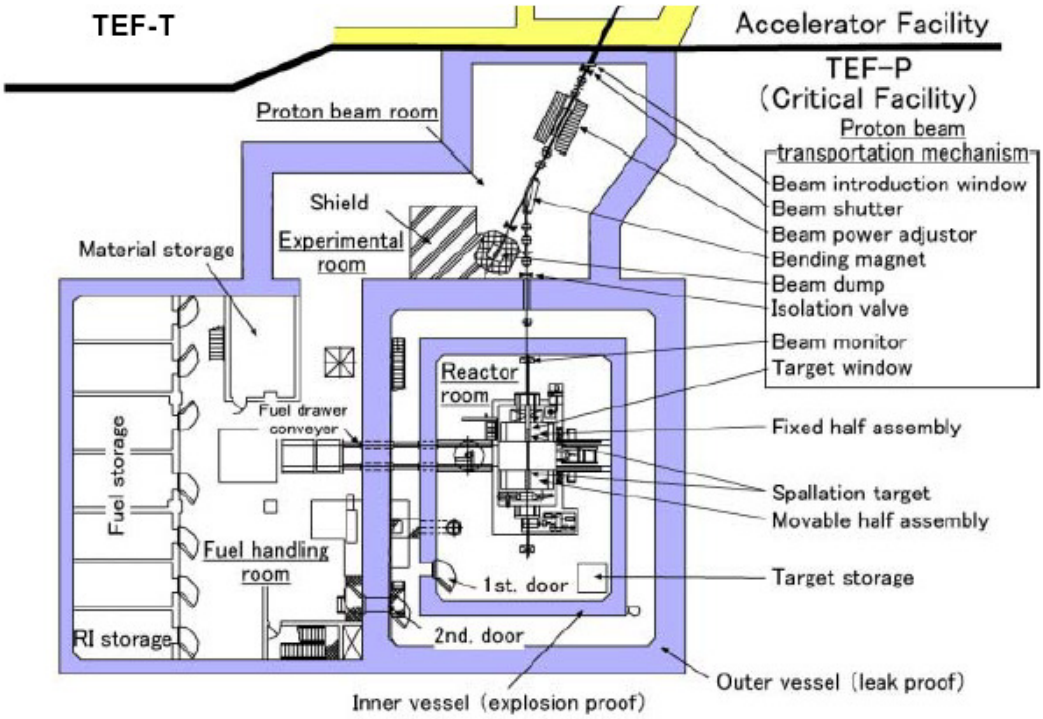

FIG. 5.15. Conceptual design of TEF-T.

Research in the Republic of Korea

KAERI initiated the ADS research in 1997. The KAERI ADS system is called HYPER (hybrid power extraction reactor). HYPER research started as a 10-year nuclear research

programme funded by the government. The ADS research of KAERI consisted of 3 stages. A basic concept of HYPER was established in the first stage (1997–2000) of the development. The basic technology related to HYPER was investigated in the second stage (2001–2003) while upgrading the design. The conceptual design of HYPER was completed in the third stage (2004–2006). The investigation of key technologies was also continued in the third stage.

The conceptual design of HYPER core was almost finished in the second stage. The upgrade of core design and transient study was done in the third stage. Regarding experimental research, a fuel and Pb–Bi coolant study were performed during the second and third stage. U surrogate fuel was fabricated and tested. KAERI joined the MEGAPIE project in 2001 for Pb–Bi research. KAERI also installed a static Pb–Bi corrosion test device in 2003 (see also Section 2.5.6). KAERI launched an I-NERI project in 2004 with LANL as the US partner to study the lead alloy corrosion. The planned period of the I-NERI project was 3 years.

5.4.2.3. ADS/P&T research in the Russian Federation

The main objectives of ADS development in the Russian Federation (Refs [133–135]) are as follows:

- Developing high rate linear proton accelerators with a current of 100 mA (in the range of 1000–1500 MeV) and a continuous mode of operation; a potential use of proton cyclotrons and electron accelerators has also been considered.
- Choosing a target for transformation of the flux energy into the neutron flux; at present, solid-phase and liquid-phase neutron-producing targets are considered in the Russian Federation.
- Selection and development of a subcritical blanket.

The major attention is focused on the investigation of cascade (sectionalized) blankets.

In the cascade subcritical liquid salt reactor (RRC Kurchatov Institute) [136] there are two breeding cores. In the first core, the neutrons have a fast or intermediate fission spectrum while in the second, core thermal neutrons are used.

Under consideration are both a homogeneous option with transmutation of neptunium with production of mixed fuel (oxides, nitrides, alloys, etc.) and a heterogeneous option with the use of special targets with uranium, inert (MgO-NpO_2–50%, $MgAl_2O_4$-NpO_2) and other matrices.

The heterogeneous method enables an option of neptunium combustion with production of ^{238}Pu, which is the most long lived nuclide as compared to neptunium ($2.14\ 10^6$ and 87.7 years, respectively) and can be used in national economy for manufacturing of radioactive heat sources. A PUREX process version has been implemented at the RT-1 plant with separation of plutonium that can be used for production of ^{238}Pu or sent to MOX fuel fabrication facilities. The existing MOX fuel manufacturing technologies require a certain upgrade for obtaining a triple U–Np–Pu composition.

5.4.2.4. ADS/P&T research in the USA

The ADS efforts are dedicated to the Accelerator Transmutation of Waste (ATW). This effort is performed within the Advanced Fuel Cycle Initiative (AFCI)[23]. This initiative was launched

[23] See also Section 3.4.4.

by the US DOE in fiscal year 2003 as an outgrowth of the Advanced Accelerator Applications (AAA) Programme, which was authorized by the US Congress in fiscal year 2001, mainly to address nuclear energy and waste management concerns (Refs [137, 138]). This effort will address long term issues associated with SNF, specifically the development of fuel cycle technologies that could sharply reduce the long term radiotoxicity and long term heat load of high level waste sent to a geologic repository.

A roadmap study, mandated to the US DOE by the US Congress, has been completed. This study describes a R&D programme recommended for federal support. It identifies the challenges of ATW concept, the associated R&D and preliminary budgetary needs, as well as potential spin-offs of this R&D. Both the roadmap document and the indications put forward by the US Congress are emphasizing very strongly the need and importance of international collaboration.

The situation in this area was presented and discussed in an IAEA conference in 2003 (IAEA International Conference on Innovative Technologies for Nuclear Fuel Cycles and Nuclear Power, held on 23–26 June 2003 in Vienna, Austria).

Today, one can say that a considerable amount of work has already been done. Accelerator equipment is being developed separately and it seems possible to obtain the required parameters. The technology of the secondary circuit should be rather similar to that used presently. However, still many different conceptual designs considering different spectra, fuels, fuel cycles and coolants are being investigated.

Accelerators in the 30 MW power range are suggested to drive an ADS for energy production and/or waste transmutation. However, the highest power presently achieved is in the 1 MW range worldwide in two machines: in the 800 MeV LINAC at LANL, USA (1.3 MW), and in the 590 MeV cyclotron at PSI, Switzerland (0.89 MW).

Various experiments are underway or are planned for the future in support for ADS conceptual designs, e.g. the ISTC 559 high energy target design of a liquid Pb–Bi spallation module, the windowless design experiments at SCK̃CEN, the Pb–Bi lifting experiments at CIRCE, and KALLA at FzK, Germany.

Nevertheless, research in this field is in its early stages. Still, more theoretical work is necessary on nuclear data and on codes. A lot of work and consequently a lot of money needs to be spent to end up with commercially mature solutions. And still many, may be, too many different concepts are studied.

Therefore, even more efforts should be taken to integrate the still ongoing many national activities into international cooperations. A leading role for these efforts should be with the IAEA and its INPRO activities.

CHAPTER 6.
FINAL CONSIDERATIONS

6.1. Nuclear fuel cycle R&D perspectives and INPRO

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 INPRO 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.

This assessment could start with already existing technical experience of innovative NFCs considered, i.e. results of theoretical studies, laboratory work, pilot plant operation and perhaps semi-commercial experience with single NFC components a well as with the full NFC. It could also include the programmes and plans for future R&D activities in the area of innovative NFCs. This experience is mainly available, if at all, in national institutions and companies, and with very few exceptions in internationally funded and operated institutions like those of the European Community (e.g. the Transuranium Institute), and last but not least, in IAEA databases.

Therefore, a realistic survey on existing experience and future planned work could be obtained by a coordinated collection of relevant data from all States interested in this survey. The INPRO project provides the unique opportunity to involve all members of this project, i.e. presently (2007) 27 countries and the European Community.

For an effective cooperation of INPRO Member States it is mandatory that the information to be collected would be structured in a way that makes the information obtained from different sources comparable. That means that as a first step INPRO should provide specifications of the kind and structure of the information to be provided by the individual countries participating in this project.

The next step then could be the evaluation of the received information, again in a way to be organized by INPRO. The target of this evaluation could be to provide an overview on existing R&D experience and a plan for the future that enables the participating countries to identify possibilities for joint R&D efforts or at least cooperation between these countries that improve the effectiveness of the efforts in each country.

By this approach, in particular developing countries could make the best out of their resources and get a chance to participate at the front of the international development.

6.2. Specific needs of developing countries

Typically developing countries are short in capital and have only limited resources of manpower and equipment to establish nuclear energy capacities. In most cases they not only cannot afford big installations, e.g. large size nuclear power plants with optimal electricity production cost, but they also do not need or cannot handle those big plants due to a low transmission capacity or size of their electrical grids.

Further, it is not only in the interest of the individual developing country to get access to technologies that are optimal with regard to economics as well as to safety and proliferation resistance and also do not unnecessarily harm their environment, but also the international community as a whole should have a vital interest to help the developing countries to establish a nuclear energy technology that is optimized in regard to the mentioned criteria.

In addition, of course, each developing country will be interested to use its national natural resources like uranium-, thorium- or zirconium-ore. Therefore, the approach to nuclear energy will be different from one country to the other.

However, some principles may be always observed. First of all, any establishment of nuclear energy in a developing country must start with a technology that has shown its maturity elsewhere. That holds not only with regard to the reactor technology but also to any other technology of interest for this country, e.g. mining, milling and conversion technologies in case they want to use their own raw materials. And if a nuclear installation with mature technology is purchased from a developed country it must be assured that also the experience to run this installation economically, securely and safely will be obtained together with the installation.

There is hardly a way for any developing country to get around these basic rules before the respective country may start efforts to participate in the development of innovative nuclear technologies. But, as already mentioned in the previous sections, there should be a strong interest in the development of innovative nuclear technologies, as well in the developing countries (technology users) as in those members of the nuclear community with large nuclear experience (technology holders). The latter should take any effort to help the developing countries to further increase their experience via participating in the development of advanced technologies according to their capabilities. The INPRO project intends to play a key role to facilitate this approach.

Some ideas may be mentioned on how this approach could work in reality. Many developing countries so far could not afford to deploy nuclear power plants. However, some of them are operating nuclear laboratories, and one or more research reactors. Those research reactors could be used to investigate specific R&D topics relevant for innovative NFC, e.g. research being performed in such nuclear laboratories and test reactors using UO_2 coated particles (TRISO-type) would be useful for every country interested in HTR technology. If technology holder countries would outsource part of their HTR fuel R&D to such a country, this on the one side would strengthen the nuclear capability in this developing country and on the other side it would safe R&D money in outsourcing countries with high labour cost.

Another big chance for developing countries could be the realization of bilateral or multilateral nuclear fuel cycle centres as recently proposed by the IAEA [142]. In Annex B, the possibilities to achieve improved proliferation resistance are discussed using such NFC centres. However, clearly there are also other advantages. One is the possibility to provide access to NFC back end technologies to those countries that cannot afford their own development. But much more important is the chance for many countries to get access to NFC back end facilities thus becoming able to treat their own spent fuel and possibly even get rid of the burden of high level waste if they do not have adequate geological conditions within their own borders.

6.3. Legal aspects

Innovative nuclear fuel cycles may — and should — provide attractive new technical and economical perspectives with also new approaches for more (passive) safety and improved proliferation resistance. Nevertheless, any new technology also needs to get licensed. As far as we can look into the future this will always be a national task, or in some cases, like in the European Community, one day a regional task.

The basis for any licence is the respective laws providing the legal outlines. These legal outlines are not a static feature. As time goes on, they will generally change most probably into two directions:

- tightening the requirements for safety and the protection of the environment according to the ongoing national political sensitivity to these aspects;
- harmonizing internationally laws due to ongoing pressure not only in the area of technologies but also in other areas with increasing international importance, as e.g. laws to fight against international terrorism.

Any effort to develop innovative technologies, and in particular innovative nuclear fuel cycles, needs to monitor these legal developments and has to take care of them in the earliest possible stages of their conceptual development. Since there is, at least, an obvious necessity to more and more cooperate internationally, if not to really internationalize the R&D for innovative nuclear fuel cycles, there is also a need to develop an internationally harmonized legal framework for the respective technologies.

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ANNEX A
RUSSIAN CONSIDERATIONS OF A GLOBAL AND/OR REGIONAL INNOVATIVE NUCLEAR ENERGY SYSTEMS

Firstly, the global situation of nuclear energy will be analyzed, and secondly the situation in the Russian Federation.

A1. Global nuclear energy system

A1.1. Introduction

International cooperation factors are presented in Refs [A-1 to A-12], which promote the efficient development of nuclear power in various regions of the world within the framework of the global energy system. These factors include:

- optimization of the nuclear power structure in various regions and countries, as a part of an international multicomponent system, by considering the capacities and facilities it consists of;
- accounting of the energy demand requirements and peculiarities in various countries with real need of nuclear power and various infrastructure development levels;
- accounting of the impact of political, economic and environmental problems in various regions of the world;
- possibilities and advantages of creating international fuel cycle centres from the viewpoint of environmental efficiency, ecological requirements, safety ensuring and prevention of illegal use of nuclear materials in military programmes;
- international harmonization of licensing and regulatory procedures;
- optimization of intellectual resources' use by developing international cooperation in R&D, and preserving knowledge and experience by improving multinational structures in the field of education and development.

At present the possibilities of creating an international nuclear power (NP) infrastructure should be considered on a quantitative level. In order to do that, it is necessary to propose a model of the global nuclear energy system (NES) taking into account the specifics of the regional development of the world and changing with time (fifty to hundred years). This model should include the necessary elaboration of details (structure, capacity range, application areas, etc.), which would make it possible to test the INPRO methodology on its base, and issue recommendations on its improvement and on the further development of the project.

As such a regional model of the world, the present analysis uses the distribution of countries adopted by the IAEA, which reflects their purely geographical position (Fig. A.1).

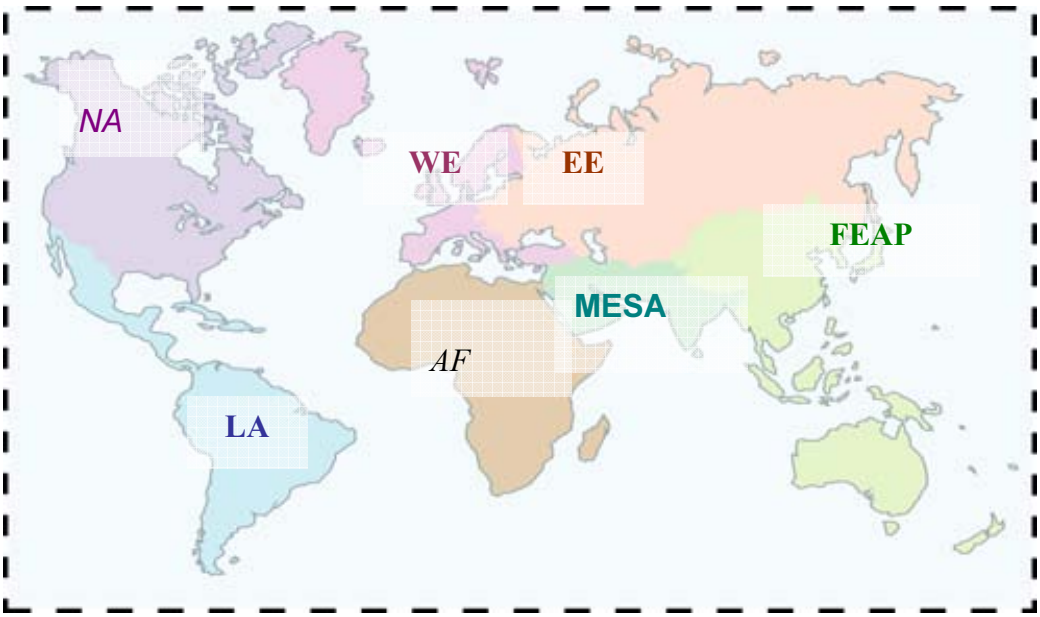


FIG. A.1. Definition of world regions.

A1.2. Initial conditions for selecting the parameters of the global nuclear energy system model

The choice of the input parameters for the global nuclear energy system model should be based on the projections of the world energy consumption dynamics, which is determined by the forthcoming demographic growth and the rates of the global and regional economic development. In accordance with INPRO recommendations, it was proposed to base these projections on the results of the study carried out by a large international group of experts in emission scenarios (SRES), the results of which were published in 2000. This group has prepared, under the auspices of the Intergovernmental Panel on Climate Change (IPCC), the well-known 40 scenarios (divided into four major groups) for demographic growth, economy and energy demand development rates, as well as for energy resources' production structure. Taking into account their international authorship and consideration by governmental and scientific experts, the SRES scenarios can be acknowledged as the state of the art in the field of long term energy forecasts.

A2 and B2 scenarios, despite their difference in global parameters, provide for close and average for the group of 4 scenarios — but relatively high — rates of development of the world nuclear energy system. The amount of nuclear energy produced in the world, according to A2 and B2 scenarios, makes about 6% of primary energy by 2050, and about 12% — by the end of the century (for the method of converting nuclear electricity into primary energy adopted in), that approximately corresponds to NPP capacity of 2000 GW and 5000 GW, respectively. For these scenarios, the rate of nuclear electricity in the total electricity production makes 25–30% in 2050, and 30–50% in 2100.

As it will be shown below, the present work analyzes the scope of nuclear energy system capacities, which satisfies A2, B2 (and naturally, B1) scenarios, taking into account additional possibilities of the aggressive nuclear power, which increases its market attractiveness by partially replacing other energy sources — both in electricity generation, and in such areas as hydrogen, heat and potable water production.

This diapason also includes the original AIT scenario, but it doesn't reach the level of the aggressive version proposed in, which cannot be realized by the world nuclear power under resource and technological constraints discussed below.

It should be emphasized that the energy scale (EJ) transformation into the scale of nuclear electric capacities (GW(e); 7000 hours of NPP operation at installed capacity), assumed in this work, was adopted only for illustrative purposes, and doesn't mean that the nuclear electricity is used only for electricity generation — moreover, the consumption structure was not considered in the current analysis (excluding the specially mentioned cases).

A1.3. Mathematical model of an energy system development

The model is intended for quantitatively assessing the dynamics of changes of the main nuclear energy system indicators, depending on the scenarios of development of innovative technologies, reactors and fuel cycles:

Time scale

- pre-history of power development: 1950–2000;
- calculated projections: 2000–2050;
- assessment of after-effect: 2050–2100.

Territory

The mathematical algorithms and service arrangement of the model make it possible to perform calculations both on the regional level (for the above groups of countries) and for the whole world.

Model contents

The general scheme of the mathematical model reflects the dynamics of changes in the global and/or regional power complex structure, including extraction and consumption of various energy resources (gas, oil, coal, nuclear energy, renewable sources), as well as non-traditional energy sources. In parallel, the material and financial balances are calculated.

Nuclear power is represented both by the existing reactors of various types, and by the innovative reactor technologies (reactors and fuel cycles) integrated in the systems, with the rate and time schedule determined by user. The isotope balance of nuclear power — with account of fuel cycle closing and assessment of the most important factors of fuel use — is also calculated.

The criteria, important for ensuring stable and safe nuclear power development, reflect the issues of fuel supply, radioactive waste management and non-proliferation. It is assumed that some of the regions will not possess the whole fuel management chain (enrichment and reprocessing), but will be using the foreign fuel ready for use in the reactors. i.e. the possibility of establishing the international fuel management centres is considered, along with the possibility of supplying non-nuclear products produces in other countries, to some regions — for example, supplying hydrogen as an energy carrier.

The mathematical model calculates the scale of material flows related to the movements of fresh and irradiated fuel; estimates the necessary scale of reprocessing and disposal of fission products, and makes assessment of the possible minor actinides management methods.

All calculations are accompanied with economic assessments.

A1.4. Versions of a global nuclear energy system

One of the key factors determining the possible look of a global nuclear energy system of the XXI century is the current estimation of the world uranium resources.

The most widely used uranium resource estimations are contained in the Red Book periodically published by NEA/OECD — a summary of the world data on uranium resources, production and demand. According to 2001 overview, the well-known uranium resources of the world (of reasonable assured resources (RAR) and estimated assured resources (EAR) categories), which have the highest price considered today — US $130/kg of U, additional resources estimations-based rather on the trends and similarity characteristics, theoretical resources estimated based on geological extrapolations, and, finally, resources included into expert considerations without any cost estimations yet, give the total of 16.2 million of metric tonnes of U.

Inclusion of the already extracted resources — commercial reserves, military material stocks, and possible depleted uranium enrichment — could increase the estimated amount of the world uranium resources up to 17.1 million of metric tonnes.

The criticism of this amount as an upper estimation of the world uranium resources — contained, for instance, in the report published by the Harvard University expert group — seems to be well grounded. The low level of the estimated resources is explained by insufficiently active financing of uranium deposits' exploration, absence of incentive for looking for uranium of higher price categories than the current US $20–40/kgU prices, the possibility of producing uranium as by-product of other resources, and to some other considerations.

However, the higher estimation of the amounts of uranium available for extraction at a cost of US $130/kgU and over, are based either on qualitative considerations, or on long term extrapolations of the correlation between the cost and the available resources (Fig. A.2), which is more or less known for the price below US $40–50/kgU. Nevertheless, use of the amounts between 20 and 40 million tonnes for the understanding of nuclear energy look's sensibility to the growth of uranium resources seems reasonable enough. But, it also requires more detailed consideration of the ecological factors, which could become important at such uranium production rates.

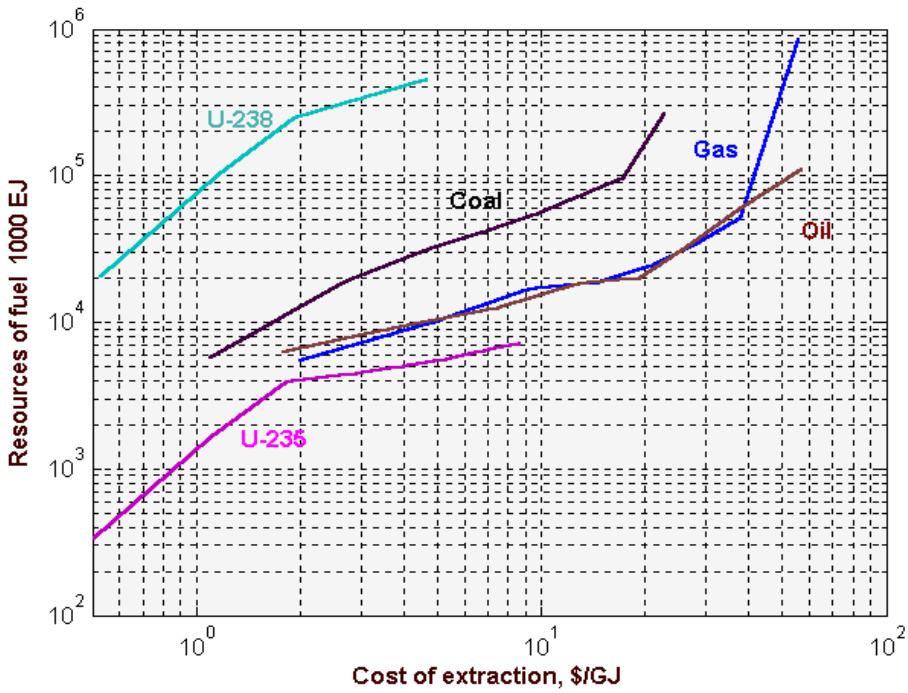


FIG. A.2. Energy resources depending on their extraction cost.

Certainly, all these estimations don't take into account the possibilities of uranium extraction from seawater (about 4 billion tonnes), which, possibly, could become feasible in future, because the acceptable technologies for uranium extraction from seawater are being continuously searched for.

Recognizing the considerable uncertainties in the assessment of uranium resources and costs, and understanding the principal impact of this data on the development scale, structure and technological development of nuclear energy, we should recommend a special research dedicated to this issue.

In this connection, the statement of the MIT Report, urging the US government to "implement the programme of assessing the world uranium resources, including geological prospecting, in order to determine the world uranium resource base with higher degree of reliability" deserves full support.

It should be noted that thorium could provide a several-fold extension of the nuclear power fuel base — but this would require the creation of thorium extraction, production and processing industry. Thorium, as a potential fuel resource, doesn't compete with uranium, but helps to solve some nuclear energy problems: minimizing the amount of transuranic nuclides in the system, and extension of the possibilities of using the reactors with thermal and intermediate neutron spectra — for instance, the high temperature heat and hydrogen cogeneration reactors. However, during the period of rapidly growing NP capacities, large scale introduction of thorium in NP would hamper its capacity growth, because the double time of the nuclear fuel in thorium-based system is much longer than in fast reactor systems operating on uranium/plutonium fuel. Use of thorium in the global nuclear energy system was not considered in the current report, and the estimation of its contribution to the nuclear energy fuel base requires a special study.

A1.4.1. Open fuel cycle system constraints

Use of LWRs[31] in the open fuel cycle, in accordance with nuclear power capacity requirements under A2 and B2 scenarios, would lead to high natural uranium consumption rate (Fig. A.3). For example, if the system's capacity increases to ~2000 GW(e) by 2050, the annual uranium extraction should reach the level of over 300 thousand tonnes, and the integral uranium consumption by 2050 could make over 10 million tonnes. By 2100, if the system's capacity increases up to 5000 GW(e), the annual uranium consumption would reach approximately 800 thousand tonnes (with close separation industry capacity), and the integral uranium consumption would exceed 40 million tonnes. The integral amount of SNF would exceed 3500 tonnes (over 50 storage facilities of Yucca Mountain capacity).

[31] Consideration of only LWRs as thermal neutron reactors, assumed in the present work, is also a conditionality related to the need of using specific reactor parameters in the calculations. Consideration of other reactors (HTGR, CANDU, etc.) would complicate the calculations reflecting notably on the main results.

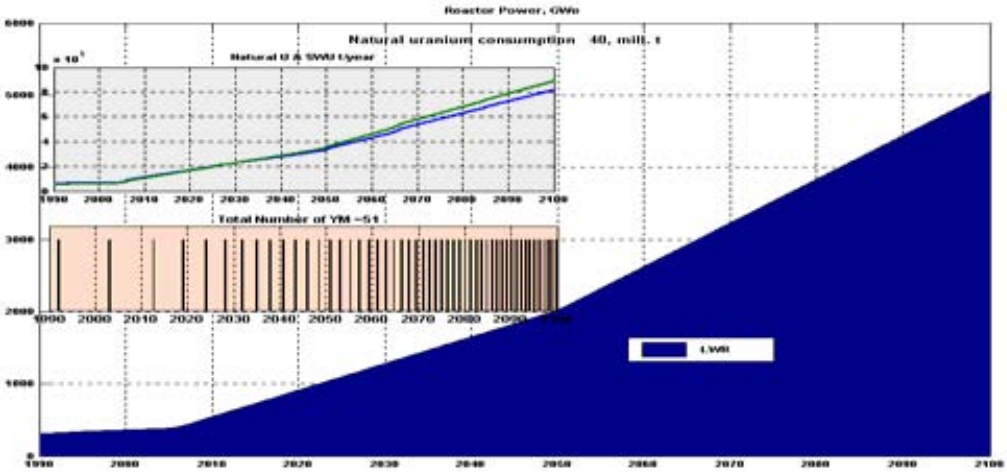


FIG. A.3. LWR-based nuclear energy system.

Thus, the nuclear energy system with once-through fuel cycle, even for the considered moderate scenarios, faces constraints related to estimated uranium resources. Development of uranium extraction and enrichment of such scale, in all probability, would require not only the expansion of the existing production capacities, but also the development of new technologies, which would solve — among others — the environmental issues.

A1.4.2. Constraints of the system without plutonium breeding

Closing the light water reactor fuel cycle with extracting Pu and using it for loading into fast reactors without breeding (BR~1.06) also provides no possibilities of reaching the capacity levels proposed for NP under A2 and B2 scenarios (Fig. A.4). If about 14 million tonnes of natural uranium are used, the capacity of LWRs would reach about 1200 GW(e) by 2050, and then would reduce to zero by 2100. Total NES's capacity would reach its peak of 2300 GW(e) around 2060; reduce to 1600 GW(e) by 2100 (only plutonium fast reactors are introduced); and then the NES capacity would slowly start growing thanks to some excess plutonium produced in fast reactors (FR). After 2100, the extraction of natural uranium would stop completely. The maximum uranium extraction of 200 thousand tonnes per year would be reached by 2040, the peak capacity of separation facilities (290 million SWU) — also by 2040.

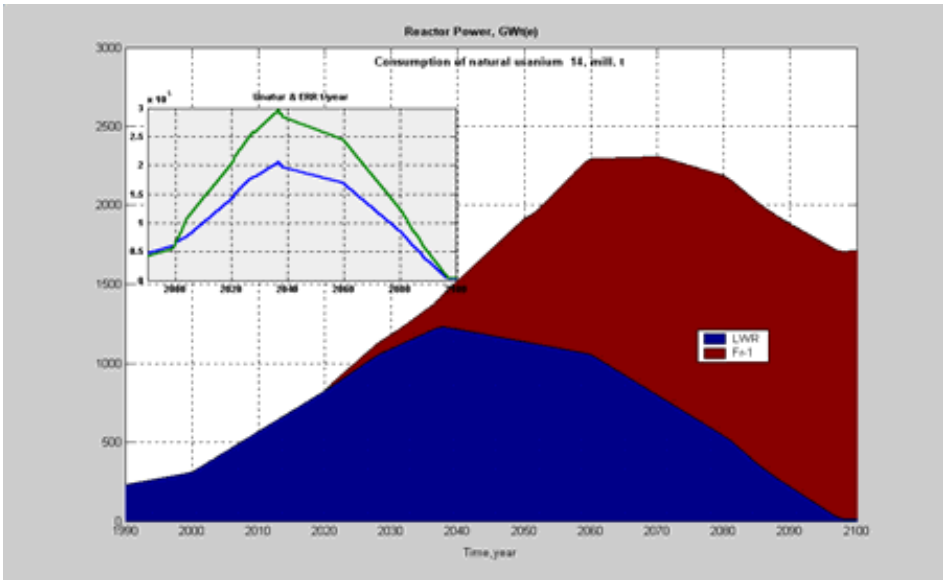


FIG. A.4. Nuclear energy system based on LWR and FR without breeding (BR~1.06).

A1.4.3. Two-component structure of NES with plutonium breeding

Introduction of FRs with plutonium breeding (BR~1.6) in the nuclear energy system would make it possible to ensure the electricity production required by A2 and B2 scenarios, without exceeding the natural uranium extraction limit of 20 million tonnes (Fig. A.5). Plutonium use starts already in 2010. The integral natural uranium extraction would make 18 million tonnes. The rate of FRs would make over 40% by 2100.

The considered two-component NES structure allows including in the consideration the aggressive nuclear energy, as well as increasing the capacity to 9400 GW(e) by 2100 (additional requirements under A2 and B2 scenarios), assuming the target of not going beyond the limit of 15 million tonnes of natural uranium per year.

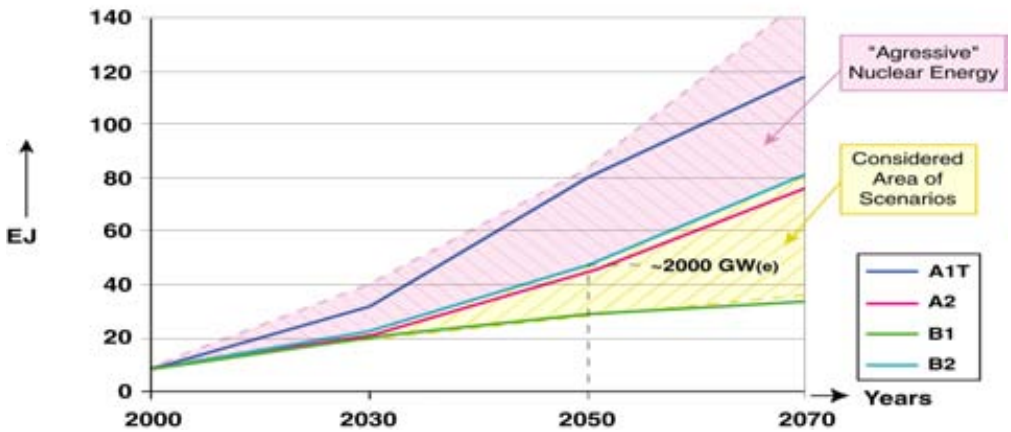
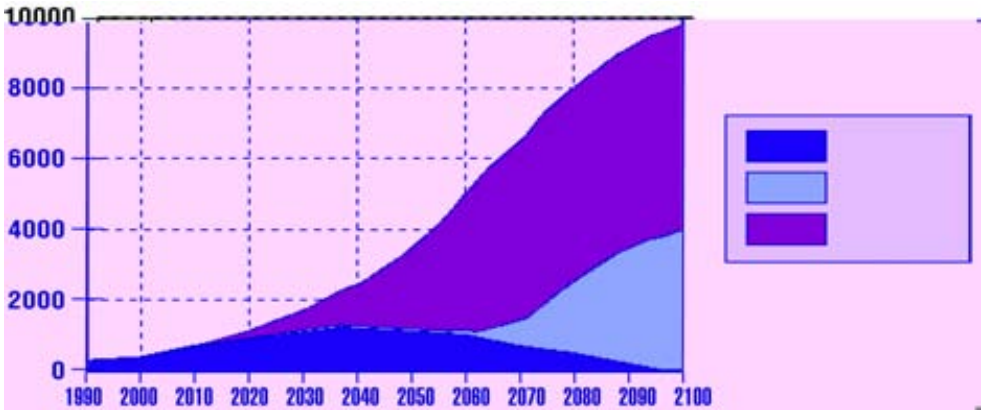


FIG. A.5. Nuclear electricity production for the four main SRES scenarios.

Plutonium utilization should be started from 2100 using the FRs from the very beginning. The mass use of MOX fuel in LWRs could not start before 2060 (Fig. A.6).



Consumption of uranium, 14 million t

FIG. A.6. 3500 GW(e) by 2050; NES based on FBR, LWR, LWR-MOX.

A1.5. Conclusions for the global nuclear energy system model

Consequently, based on the present day estimated uranium resource limitations, the realization of the given nuclear power development scenarios (Chapter 1) — and, moreover, additional ones — in practice leave us the two-component structure of NES with Pu breeding for further consideration. Obviously, such a forecast makes the search of additional uranium resources an actual task.

Besides, it is easy to see that these so-called moderate scenarios of the world nuclear energy development, which nevertheless allow growth up to about 10000 GW by the end of the century for producing, for example, nuclear hydrogen, suppose strict enough conditions for the rates of implementing the technological innovations, aimed at solving the problem of uranium resource constraints. These innovations, which ensure improved fuel use, include the closed furl cycle with short reprocessing time, very good breeders (with BR~1.6) and very good LWRs with BR~0.9 using uranium–plutonium fuel (Fig. A.7). It should be noted that it is hard to imagine such a high rate of innovations' implementation without special governmental (and intergovernmental) support. While preparing corresponding governmental decisions, it is necessary to take into account the inertia of nuclear technology development. That means that, in a critical energy resource situation, even governmental decisions will be unable to make nuclear energy instantly provide and implement new technologies.

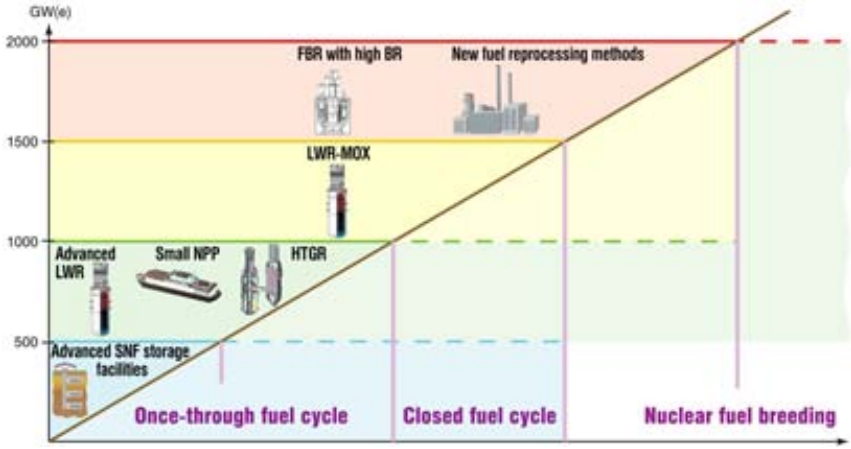


FIG. A.7. Required innovations depending on the level of the world nuclear energy development (by 2050).

Consequently, governmental institutions should ensure the advanced development of nuclear innovations. High nuclear energy development rates would require not only the development and mastering of the above innovations, but also the active search for additional uranium resources and development of new technologies of their extraction. It is necessary to emphasize that the works on searching for additional uranium resources and the innovative improved fuel use technologies don't exclude, but mutually add to each other, thus mitigating each other's urgency.

Reduction of the time of fuel stay in the external fuel cycle is a very important direction of works aimed at increasing the rate of large scale nuclear energy development. In this connection, a cardinal innovative solution is presented by introduction of liquid fuel reactors with continuous fuel processing (molten salt type), but their large scale deployment, in all probability, will go beyond the 21st century frames, and will reach its maximum efficiency at the stage of introduction of thorium in the NES as an efficient means of extending the NE fuel base, reducing the environmental impact levels, and solving the issue of minor actinides' disposal, in period of NE structure's stabilization.

Reduction of the external fuel cycle time in FRs from 3 to 1 years would make it possible to reduce fuel load in NFC (two-component NE structure) by about 15 to 25%, thus

correspondingly increasing the NE development rates[32]. Earlier, the methods of increasing the energy density of breeder reactor cores in order to increase the rate of achieving the double NE capacity have been considered, but they do too obviously contradict to the present day trends of using the intrinsic safety features and passive means for improving reliability and safety of reactor operation.

Assumed parameters, which determine the rates of Pu breeding (BR~1.6, external fuel cycle time — 3 years) allows assessing the maximum realistic growth of the nuclear energy (in case of slight mitigation of uranium resource constraints — 10 million tonnes over the assessment adopted today). Under this condition, the world nuclear power might reach the level of 12000 GW(e) (or about 300 EJ) by 2100 (Fig. A.8). In such a case, the nuclear energy system could generate (under A2 and B2 scenarios) about 40% of electricity by 2050, and about 70% by 2100[33].

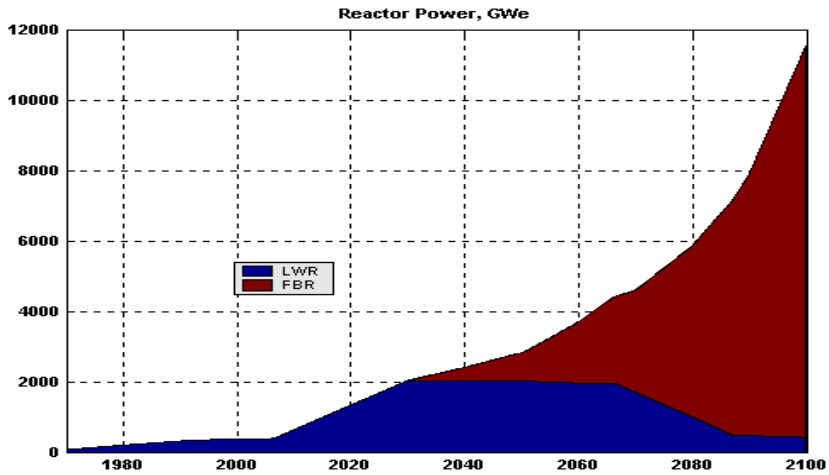


FIG. A.8. Maximum NES development scenario, based on the increased natural uranium resource estimation.

Estimations of the continued delay of development of the nuclear energy by the mid-century show that under this scenario it would be also possible to achieve the 5000 GW level by the end of the century, but this would be possible only under very strict conditions of rapid deployment of breeders and additional borrowing of about 10 million tonnes of uranium (Fig. A.9).

[32] Probably, it would be expedient to consider — as a means of reducing the nuclear fuel load in the NFC — SNF reprocessing at NPP sites with high capacity fast reactors (impact of such technologies on non-proliferation regime requires a separate discussion). But, this way would be efficient only in case of SNF unloading from reactor without it's cooling in internal reactor storage facility, and in case the fuel reprocessing technology with high residual energy and radioactivity levels is created.

[33] It should be noted that, in case of such nuclear energy development, fossil fuel extraction and carbon dioxide emissions could be stabilized on their present day levels.

FIG. A.9. Impact of delayed fast reactor deployment on nuclear power development.

It should be noted that, for example, a two-fold reduction of the nuclear power development rates (which corresponds to B1 scenario) would allow the existence of a nuclear energy system without introducing FRs with integral uranium consumption of 17 million tonnes till 2100.

Consideration of intermediate once-through cycle versions (NE system capacity of 1600 GW by 2050) results in the growth of integral uranium consumption by 2100 up to the same speculative level of about 27 million tonnes.

In the same time, the low scenario doesn't create any significant innovation problems, at least in the technology area. In the field of reactor technology development, the advancement of light water reactors (and extending their capacity range to small and medium power), as well as wide implementation of high temperature reactors for hydrogen production, would be enough.

It should be noted that the range below this scenario (1000 GW by 2050) leaves nuclear energy rather a technological demonstration, which would be an insurance against possible limitations in the development of other energy technologies.

It is seen that NE development prospects largely depend on the selected NFC. However, in present conditions, the fuel component of the energy cost, as well as today's risks related to energy generation, don't depend strongly on the fact, if SNF is stored without reprocessing, or SNF is reprocessed and only radwaste is stored. The initial stage of the nuclear fuel cycle is practically independent from the fuel cycle type, if the NE system comprises no fast neutron reactors capable of nuclear fuel breeding.

If a closed NFC is used, and if NES uses fast neutron reactors capable to utilize efficiently over 70% of uranium, the energy potential of the nuclear industry becomes much higher, than all the fossil fuel resources. The initial NFC stage is principally enhanced thanks to reduced uranium extraction, with corresponding reduction of long term risks — to several man·Sv per each GW·year of electricity produced.

Besides, the SNF reprocessing option is more economically and politically realistic, even in case of using only thermal neutron reactors, because it requires considerably smaller resources to be spent for waste management in the long term perspective, which is unpredictable in terms of economic and political conditions.

In perspective, the issue of nuclear energy's acceptability is connected sooner with the confidence in organization and realization of the fuel cycle, which would guarantee large scale and long term development without steady growing resource and waste problems (that would contradict to the Basic Principles and purposes of sustainable development), than with current capital investments in nuclear power plants.

SNF reprocessing and returning of uranium and plutonium in reactors represent a necessary condition of nuclear fuel breeding and increasing the efficiency of uranium use by 10–20% in thermal neutron reactors (and by order of magnitude — is NES uses fast neutron reactors).

Moreover, NFC closing by uranium and plutonium allows principal reduction of high level waste, which is subject to disposal. NFC closing by americium, curium and neptunium (minor actinides' transmutation in the present-day notion) makes possible a considerable reduction of risk related to long term consequences of radwaste disposal.

For efficient solution of sustainable development problems, the level of actinide losses during reprocessing should be kept on the level of about 0.1%. In the opposite case, minor actinides' transmutation — which requires additional complication of SNF reprocessing methods, reactor construction and NE structure — has no sense, because it doesn't lead to any considerable reduction of long term environmental risks, but on the contrary, considerably increases the current risk level because of growing balanced amounts of minor actinides in the reactors and NFC.

For NFC closing by neptunium and americium (and, after 2100, curium), and some other hazardous nuclides, after 2050, the introduction of liquid-fuel reactors for minor actinides' burning could be considered — but this is proposed just in case the problem of their safe and socially acceptable disposal (and, possibly, their partial use for the purposes not related to large scale energy production) is not solved.

Results of this report's analysis do not take these reactors in consideration, but there is nevertheless a possibility of checking the acceptability of such a multi-component NES from the sustainable development viewpoint. This would require calculating the amount of radionuclides for all NFC stages; and assessing the neutron efficiency of nuclide composition of nuclear fuel in the system, radionuclide losses, and the capability of burning all the hazardous radionuclides at the final stage of NP development.

It should be noted that, despite of slightly differing fuel components of various NFCs, fuel cycle structures and elements might vary considerably. That means that the present day economic assessments could hardly be laid in the base of fuel cycle selection. In conditions of NFC closing inevitability (from the point of view of sustainable development on the long term), it should be kept in mind that in the next hundred years the main problem could be represented not by high level waste (its amount, risk level and energy rate is lower, than for SNF), but by low level waste — the amount of which considerably depend on SNF reprocessing methods and state of technological processes' development.

In particular, it might be that aqueous methods of SNF reprocessing would require not only some improvement, but also introduction of various non-aqueous methods of nuclide separation, in order to considerably reduce the volume of medium and low level waste, thus making NE attractive not only on the long term, but already in the nearest future.

A1.6. Nuclear energy system versions for the regional model of the world

Projecting the regional systems on the base of the selected global nuclear energy system takes into consideration the following assumptions:

- IAEA assessments by 2030, based on national forecasts and programmes;
- detailed expert knowledge of the former USSR and East European region, with account of the adopted Russian energy strategy, was applied to nuclear power development projections for this area;
- while considering the regions' development prospects, both their economic[34] realities and state of (power grids, communications, staff, etc.), and their sensibility to availability of weapon-usable nuclear materials, have been taken into consideration.

Two approximations of nuclear power capacity distribution in the 21st century's world were considered. The first one was based on the existing level of the countries' nuclear development and their declared intentions, and was comprehensively analyzed by the IAEA to 2030 (Fig. A.10). In figures and tables it is called traditional.

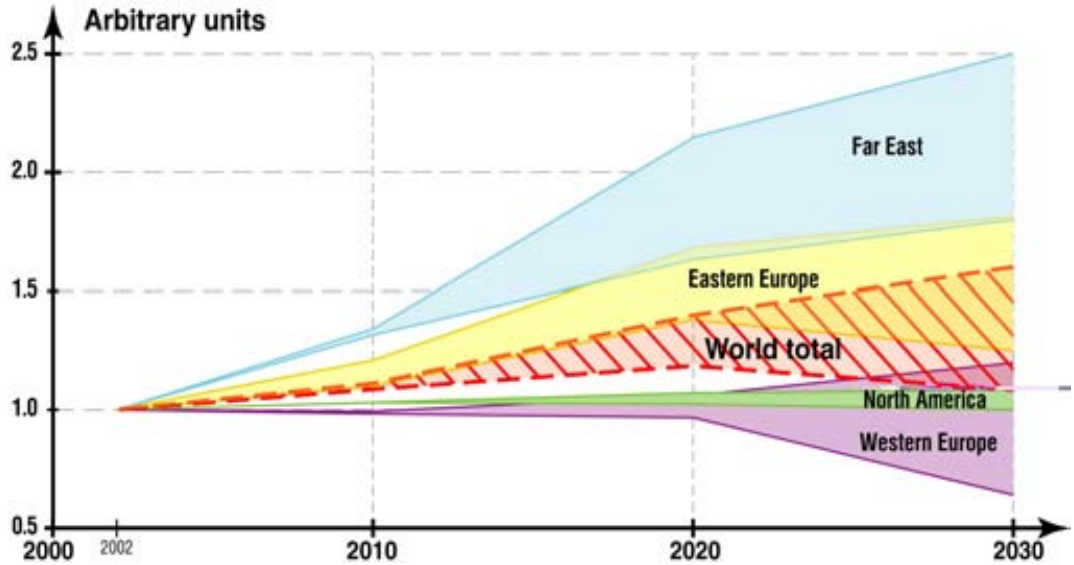


FIG. A.10. Nuclear capacity projection by the world regions (IAEA, 2003).

The second — hypothetical — approximation, which in fact represents a known convergence model, is a step to a fairer world, where the gap in the per capita energy consumption between the developed countries and the rest of the world is considerably reduced with the help of nuclear energy.

It is supposed that:

- Per capita energy consumption is levelled only through the use of nuclear energy.
- By 2040, in three regions (NA, WE, EE), annual nuclear energy consumption exceeds 4000 kW·h per capita — i.e. the world's average sufficient electricity consumption level adopted by the UN.

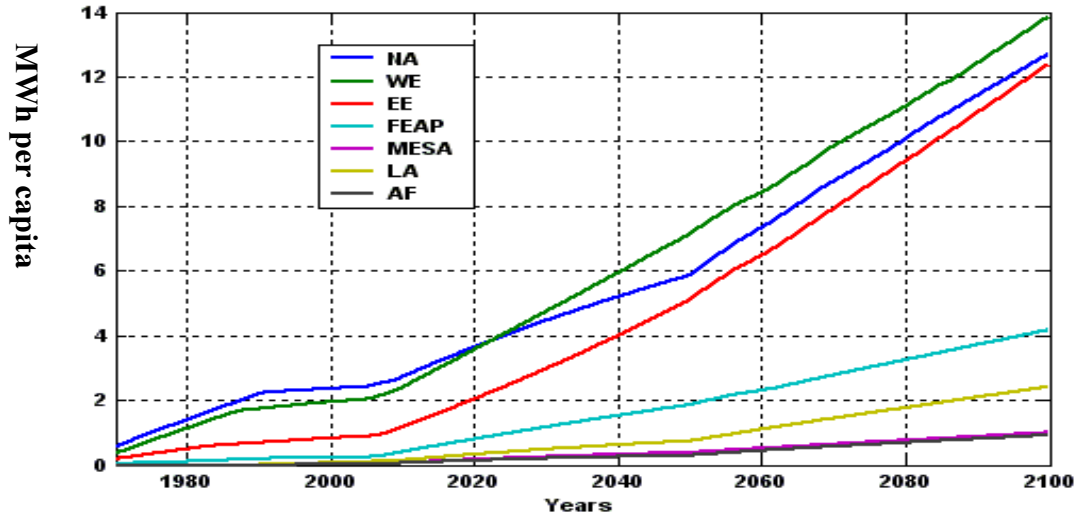
[34] According to the assessments made by Dr. H. Khatib (WEC) [8], supply of energy to the houses of 2 billion people presently living without electricity would require about US $600 billion, which is comparable to the world developing countries' assistance programme amounting to about US $100 billion/year.

That's why, since 2040, per capita nuclear electricity consumption in these regions will be stabilized (i.e. nuclear energy production will change in proportion to the population).

In the rest four regions (assuming the entry data — that the global nuclear capacity makes 2000 GW by 2050, and 5000 GW by 2100 — are preserved), the new nuclear capacities after 2040 will be growing for levelling the per capita consumption.

In such a case, FEAP and MESA regions would achieve — and LA and AF would approach to — the annual electricity consumption level of about 4000 kWe per capita (Fig. A.11).

Traditional model A



Reduced gap model B

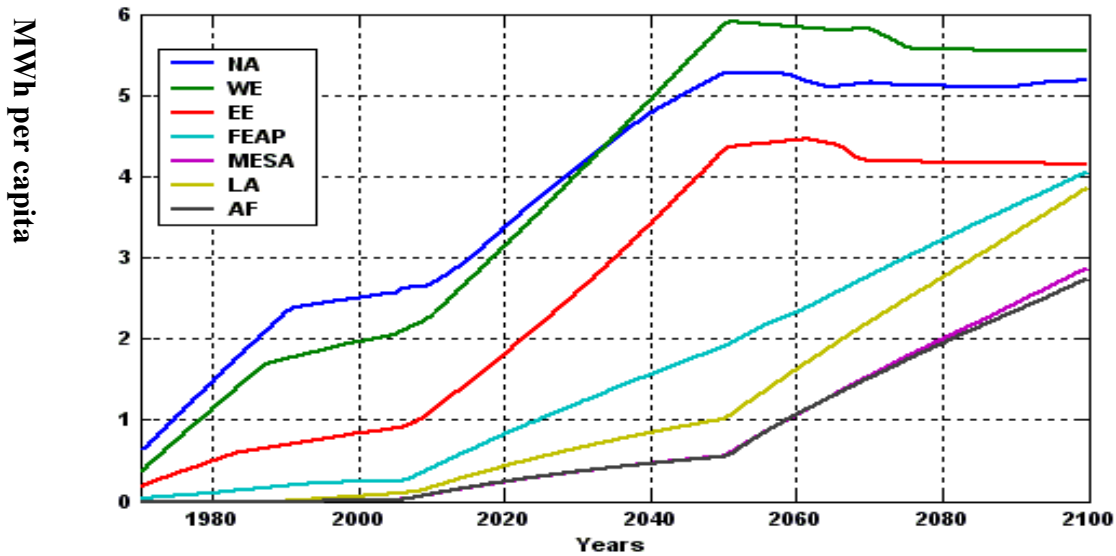


FIG. A.11. Nuclear electricity production per capita: A — traditional model; B — reduced gap model.

A2. Russian approach of a sustainable NFC

In the following text the results of a Russian study to define a sustainable development of nuclear power are laid out shortly.

Two stage concept of nuclear power development

The successful operating record of the first civilian nuclear power facilities in the Russian Federation gave rise to a concept of nuclear power development in two stages:

- Thermal reactors generate power and accumulate plutonium for the startup and concurrent deployment of fast reactors.
- Fast reactors step in later to provide large scale growth of nuclear power which will gradually replace traditional fossil energy sources.

The strategic objective was to obtain inexhaustible (sustainable) resources of cheap fuel — uranium and, possibly, converting thorium through nuclear breeding.

The tactical objective was to use ^{235}U-fuelled thermal reactors to produce power (and radioisotopes) for the national economy and to accumulate (reactor grade) plutonium for fast reactors.

Sustainability of an energy system consisting of thermal reactors

The resources of natural uranium, which may be cost-effectively produced, are limited. With the currently prevailing practice of burning uranium in thermal reactors, such resources will be exhausted within the coming century both in the Russian Federation and in the world as a whole. This time span can be somewhat extended by reprocessing spent fuel with plutonium recycling (MOX fuel) in thermal reactors — if you are prepared for increased expenses and deciding to impair the chances for future development of fast reactors.

Situation in the Russian Federation

The 1999 projections of the Energy Research Institute of the Russian Academy of Sciences point to the possibility of an increase in the nuclear electricity generation in the Russian Federation from 133 TW·h in 2003 (produced by ~22 GW(e) needing ~4700 t·U/a) up to 160 TW·h (produced by ~25 GW(e) needing 5600 t·U/a) in 2010 and up to 330 TW·h (produced by ~54 GW(e) needing 11000 t·U /a) in 2020.

According to the Red Book of the OECD and the IAEA (Version 2003), the Russian Federation's total reserves of natural uranium, produced at a cost of US $80 per kilogram, are estimated at 240000 t (RAR + EAR-I, II), while the reserves assessed with a high degree of certainty approach 150000 t (RAR). In addition to these 3 categories of uranium reserves, the Red Book refers to yet another category, SR, with the total uranium quantity of 1000000 t, including 550000 t whose production cost may be as high as US $130 per kilogram.

Over its lifetime of ~50 years, a 1 GW(e) thermal reactor (LWR) consumes ~10^4 t of natural uranium, hence 10^6 t of uranium could provide fuel for operation of 100 power units with such a reactor in the Russian Federation.

Situation globally

With the earth's population predicted to grow to almost twice its size by the mid-21st century — mostly in developing countries — and with the increasing industrialization of the latter, it is reasonable to expect doubling of the global requirements for primary energy and trebling (up to ~6000 GW(e)) of the world's electricity demands (currently ~2000 GW(e)). Nuclear power well suited to comply with safety and economy requirements of large scale power production, could take up the job of covering much of the increase in the global energy and corresponding fuel demand (~4000 GW(e)). The global uranium reserves found in the richest deposits with metal concentration in ore of 0.1% are estimated at somewhat more than 5 mio t·U for prospected resources and at 10 mio t·U for potential resources.

As stated above, over its lifetime of ~50 years, a 1 GW(e) thermal reactor (LWR) consumes ~10^4 t of natural uranium, hence 10^7 t of uranium can provide fuel for worldwide operation of 1000 power units with such a reactor; such capacity includes ~350 GW(e) provided by the plants operating today while the remaining 650 GW(e) could be brought in during the 21st century.

Evaluating demand and resources, the world's nuclear power based exclusively on thermal reactors could double its capacity to 1000 GW(e) in the first half of the 21st century, with an allowance made for decommissioning of old plants, but its contribution to energy (electricity) supply would gradually decrease to the point of total disappearance in the second half of the century.

The result of the above evaluation is:

- The existing global — and Russian — resources of natural uranium are not large enough to sustain long term development of nuclear power based solely on thermal reactors.

Sustainability of an energy system consisting of fast reactors

Uranium can be burned almost completely in a fast reactor with a breeding ratio of unity or more. A factor of 100 increase in the energy yield of nuclear fuel — as compared to thermal reactors — means that a fast reactor system of 4000 GW(e) could be supplied with cheap uranium (from rich deposits) for 2.5 thousand years, with a small contribution of fuel to the total cost, and that a 10000 GW(e) power industry can be sustained for about one thousand years. Fast reactors could be fed with uranium from lean deposits, resources of which are hundreds or even thousands of times greater than the rich uranium reserves.

The result of the above evaluation is:

- Fast reactors with a moderate power density and a breeding ratio of about unity allow large scale nuclear power development unconstrained by availability of fuel resources.

Sustainability of an energy system with a mix of thermal and fast reactors

The concept of nuclear power development in two stages suggests long coexistence of ^{235}U-fuelled thermal reactors — for as long as cheap uranium is available — with fast reactors, which will be brought in to use plutonium from weapons inventories and from thermal reactors and will have virtually no fuel constraints.

Thermal reactors included in such a binary mix would benefit from gradually switching over to the more expedient Th–U cycle, with ^{233}U for the first cores and subsequent makeup provided by Th-blankets of fast reactors. There are good reasons for the nuclear power industry of the future to have a binary structure, but the essential question of the proportion between fast and thermal reactors is yet to be adequately resolved.

Given a moderate rate of installed capacity growth, Russian nuclear power industry will remain practically uniform in the next few decades, with an insignificant energy input from fast reactors. However, in case of a rapid ascent, the nuclear industry will have to rely primarily on fast reactors, as the fuel resources available for operation of thermal reactors in the Russian Federation are not sufficient to provide a steady growth of installed capacity (i.e. 1–2 GW/a) and would be exhausted before the middle of the 21st century.

Primarily, fast reactors with a breeding ratio equal to or greater than unity can serve the purposes of large scale nuclear power. Under the current conditions of the industry's evolution (moderate growth rate), there is no need for large breeding ratios, high power densities or short plutonium doubling times, which allows focusing on economy and safety in the fast reactor development programmes.

A surplus of neutrons and their energy spectrum enabling fission of all actinides are the facts that allow fast reactors to burn effectively the most hazardous and longest lived radionuclides in the fuel cycle wastes, which paves the way for a radiation balance between the buried waste and the mined uranium, established and sustained without the aid of special burner reactors up to the final stage of nuclear power development.

One of the ways to augment the resources of the nuclear power industry lies in reuse of fissile materials — the materials separated from the spent uranium–plutonium fuel — in thermal neutron reactors. It has been estimated that by closing the fuel cycle with respect to uranium, it is possible to increase the capacity of nuclear power with mono-recycling by ~13%, while closing the cycle for both uranium and plutonium could add ~17% to this capacity. In both cases, however, the operation time of the industry based on thermal reactors remains practically the same as with the open fuel cycle.

It may be inferred therefore that plutonium recycling in thermal reactors offers little promise for replenishing their fuel resources. To attain this end, in the transition period before fast reactors of the new generation are deployed, it appears worthwhile to assess the expediency of exploring and developing new uranium deposits and to look for inherent reserves of the nuclear fuel cycle in terms of uranium saving, which may be attained, for instance, by reprocessing the enrichment tails, using regenerated and weapons uranium, etc.

Options to significantly increase safety and economy of fast neutron reactors include the adoption of nitride fuel, elimination of uranium blankets, and increase of fuel burnup.

Waste management

Closing of the nuclear fuel cycle is a strategic line of nuclear power development in the Russian Federation, which will:

- ensure more efficient use of natural nuclear fuel and artificial fissile materials produced by reactors (e.g. plutonium);
- minimize radwaste from fuel reprocessing;
- help approach a radiation equivalence between the buried waste and the mined natural fuel.

The quantities of irradiated fuel are quite large in the world as a whole and in this country, with ~250 thousand tonnes accumulated on a global scale and ~14 thousand tonnes in the Russian Federation by the year 2000 with a total radioactivity of 5 billion Ci.

The fuel unloaded from nuclear power facilities is found mostly in storage as its reprocessing is performed on a relatively small scale. Storage of irradiated fuel is quickly becoming a real challenge: considering its increasing quantities unloaded from decommissioned power, transport and research reactors, the existing storage facilities may be expected to be full in the Russian Federation by the year 2007.

However, it appears expedient to postpone large scale reprocessing of the bulk of irradiated nuclear fuel until fast reactors of the new generation are in serial construction. This allows also avoiding further accumulation of plutonium in stockpiles, as dictated by non-proliferation considerations.

Weapons plutonium utilization is to be treated as the first stage in creating the technology of the future closed nuclear fuel cycle. Disposition of surplus weapons plutonium as well as of plutonium separated from irradiated fuel, will be effected primarily by use of mixed uranium–plutonium fuel in fast reactors which will form the basis of the future Russian large scale nuclear power industry. Limited quantities of weapons plutonium may be utilized in thermal reactors, if so required by political agreements.

Summary and conclusion of the Russian study

The above-described considerations in the Russian study lead to several conclusions:

- Without an NFC concept that strongly protects and extends the available nuclear raw material resources — uranium and/or thorium — a further large scale development of nuclear power technologies just makes no sense: not economically and not technically.
- For any global NFC concept assuring a long range resource base it is mandatory to use the breeding capabilities of fast reactors and those capabilities only can be used in a closed fuel cycle.
- There are various potential technological solutions thinkable and some of them appear to be technical feasible.
- Realistically, for a global nuclear system there is a combination of thermal and fast reactor systems necessary, i.e. a plain 100% fast reactor fuel cycle is just an abstract extrapolation of technically thinkable alternatives.
- In the mid-range future the currently used classical U (Pu)-based fuel cycle will be dominant with light or heavy water reactors producing the power.
- NFC with breeding elements supplying the long range fissile material basis must be developed in accordance with a reasonable further improvement of this classical nuclear energy technology.

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ANNEX B
MULTILATERAL NUCLEAR FUEL CYCLE CENTRES

Since the late 1970s several international studies have been performed dealing with multinational nuclear approaches (MNA), e.g. the IAEA study on regional nuclear fuel cycle centres (RFCC, 1977), the international nuclear fuel cycle evaluation (INFCE, 1980), and the expert group on international plutonium storage (IPS, 1982)[35].

The following overview of the most recent activities in this area is mostly based on Ref. [B-1].

B1. The MNA expert group

In mid-2004, the IAEA organized a group of 25 experts to identify issues and options relevant to multilateral approaches for both front and back end of the nuclear fuel cycle, to provide an overview of the policy, legal, security, economic, institutional and technological incentives and disincentives for cooperation in multilateral arrangements. In the IAEA report documenting the results of this study [B-2] two primary deciding factors were identified that dominate all assessments of multilateral nuclear approaches, namely assurance of supply and services and assurance of non-proliferation. Three types of options for creating multilateral arrangements have been presented:

- Type I: Assurances of services not involving ownership of facilities:
  - a. Suppliers provide additional assurances of supply.
  - b. International consortia of governments provide additional assurances.
  - c. IAEA-related arrangements strengthen assurances.
- Type II: Conversion of existing national facilities to multinational facilities.
- Type III: Construction of new joint facilities.

The expert group recommended to follow these suggestions:

The objective of increasing non-proliferation assurances concerning civilian NFC, while preserving assurances of supply and services around the world could be achieved through a set of gradually introduced MNAs:

(1) Reinforcing existing commercial market mechanism on a case by case basis through long term contracts and transparent suppliers' arrangements with government backing. Examples would be: fuel leasing and fuel take-back, commercial offers to store and dispose of spent fuel, and commercial fuel banks.
(2) Developing and implementing international supply guarantees with IAEA participation. Different models should be investigated, notably with the IAEA as guarantor of service supplies, e.g. as administrator of a fuel bank.
(3) Promoting voluntary conversion of existing facilities to MNAs, and pursuing them as confidence building measures, with the participation of NPT NNWS and NWS, and non-NPT States.
(4) Creating through voluntary agreements and contracts, multinational, and in particular regional, MNAs for new facilities based on joint ownership, drawing rights or co-management for front end and back end nuclear facilities, such as: U enrichment;

[35] Short descriptions of RFCC, INFCE, and IPS can be found in Annex IV of Ref. [B-2].

fuel reprocessing; and disposal and storage of spent fuel (and combinations thereof). Integrated nuclear power parks would also serve this objective.

(5) The scenario of as further expansion of nuclear energy around the world might call for the development of a NFC with stronger multilateral arrangements — by region or by continent — and broader cooperation, involving the IAEA and the international community.

After this IAEA study several follow-up proposals in regard to Type I have been made and intensively discussed.

B2. Russian initiative on international nuclear fuel cycle centres

In January 2006, an initiative was announced by the Russian president to develop a global nuclear power infrastructure providing equal access to nuclear energy for all interested countries while ensuring strict compliance with non-proliferation requirements. A network of international centres rendering IAEA safeguarded nuclear fuel cycle services, including U enrichment should become the key element of such an infrastructure.

In October 2006, the Russian Federation declassified the Angarsk enrichment facility and promoted it as International Uranium Enrichment Centre (IUEC) under the IAEA [B-3] oversight. The concept asks for investment in the facilities by potential customers who in return would receive assured supply of low enriched U. As a prerequisite, customers have to comply with the international non-proliferation requirements and forgo the development of indigenous enrichment facilities.

A similar proposal was also made by Germany in 2006, namely to build an international uranium enrichment facility at an extraterritorial site [B-4].

B3. Study of the uranium enrichment industry

In May 2006, the World Nuclear Association (WNA) released a report [B-5] documenting the results of a 28 member panel of nuclear industry experts studying an industry-based backup supply mechanism. The WNA report states that any approach strengthening security of supply should be consistent with continued effective operation of the existing competitive world market. Customers of enrichment services should pay a premium for guaranteed supply and commit to forgo building enrichment facilities themselves.

B4. The six country proposal

In June 2006, six countries [B-6] with commercial enrichment activities — the USA, the UK, France, Germany, the Netherlands and the Russian Federation — offered reliable access to nuclear fuel for States opting to rely on the international market for nuclear fuel and forgo domestic enrichment activities. The role of the IAEA is to ascertain that such States have a comprehensive safeguards agreement and an additional protocol in force and have no exceptional safeguards implementation issues outstanding.

In context of the six country proposal, in September 2006, the UK proposed the concept of Enrichment Bonds [B-7]. Such bonds would enable qualifying recipient states receiving enrichment services through the IAEA. This initiative was supported by Germany and the Netherlands. At the same time Japan proposed to establish a system called IAEA Standby Arrangements for the Assurance of Nuclear Fuel Supply [B-8].

B5. IAEA fuel reserve

In September 2006 a non-governmental organization called the Nuclear Threat Initiative (NTI) proposed [B-9] to set up a stockpile of low enriched U under the IAEA's auspices for countries to serve as a last resort fuel reserve under the condition that they do not start a national enrichment programme. The IAEA is expected to develop the modalities of such a fuel reserve regarding technical and legal issues.

Reference [B-1] provides some discussion on the initiates shortly presented above, illustrating the strengths and weaknesses. The main recommendation is to follow an incremental approach to develop a new multilateral framework for the nuclear fuel cycle that is voluntary and guarantees the States the freedom to choose their fuel options.

B6. Multinational options for the storage of spent nuclear fuel or nuclear waste

In 2002, the international conference on Issues and trends in Radioactive Waste Management [B-10] gave an overview and status of nuclear countries concerning their position of final disposal. The position of European states turned out to be different concerning the construction of international SNF storage facilities, based on different interests: France and Great Britain could see it as a competition (to their existing reprocessing facilities); Finland and Sweden have their own disposal concept without the need of reprocessing or disposal in an international repository; Belgium, the Netherlands, Spain, the Czech Republic, Slovakia, Slovenia, Bulgaria, Hungary, Romania and Switzerland could have a potential interest in international SNF storage and reprocessing, if this is a step toward an international final disposal concept for a repository in a deep geological formation and if the proposed concept is more cost effective as it is now. International SNF storage and reprocessing without final disposal would not fit the demand of these countries as they all intend to avoid extra international transports with all the risks of public acceptance (one transport to the interim storage and one return transport of waste).

No shared multinational repository exists currently [B-11]. However, several multinational initiatives are pursuing the idea of such a repository. Examples are:

- The Arius Association consists of several national organizations dealing with the storage and disposal of nuclear waste; Belgium, Bulgaria, Hungary, Italy, Latvia, the Netherlands and Slovenia participate in this initiative.
- The SAPIERR project is feasibility study supported by the European Commission investigating the option of a regional nuclear waste facility.
- The Ljubljana Initiative is carried by Austria, Bulgaria, Croatia, the Czech Republic, Hungary, Slovakia and Slovenia; the participants are assessing the potential advantages of shared solutions for nuclear waste.

The Russian Federation is the only country publicly supporting at the government level the import of spent fuel for storage.

The IAEA has continued to work on this topic with several working groups and has published two related documents (Refs [B-12, B-13]).

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ABBREVIATIONS

a	year
ADS	Accelerator Driven System
AFC	Alkaline fuel cell
AFCI	Advanced Fuel Cycle Initiative
AGR	Advanced gas reactor
AL	Acceptance limit (INPRO)
ALARP	As low as reasonably practical (social and economic factors taken into account)
BOO	Build, own and operate
BOT	Build, own and transfer
BP	Basic principle (INPRO)
BWR	Boiling water reactor
CNS	Current Nuclear System
DU	Depleted uranium
EJ	Exajoule (10^{18} Joules)
EUR	European Utility Requirements
FCF	Fuel cycle facility
FP	Fission products
FR	Fast reactor
ft	Feet
GB	Gigabarrels
gC/MJ	Grams carbon per megajoule
GDP	Gross Domestic Product
GFR	Gas cooled fast reactor
GHG	Greenhouse gas
GIF	Generation IV International Forum
GJ	Gigajoule (10^9 Joule)
GJ/s	Gigajoule per second
g/MJ	Grams per megajoule
GW	Gigawatt (10^9 Watt)
GWP	Global warming potential
HEU	Highly enriched uranium
HF	Human factor
HLW	High level waste

HTGR	High temperature gas reactor
HWR	Heavy water reactor
I&C	Instrumentation and control
ICE	Internal combustion engine
ICG	International Coordinating Group in INPRO
ICS	Individual case study (INPRO)
ICRP	International Commission on Radiological Protection
IDC	Interest during construction
IGCC	Integrated gasification combined cycle (coal power plant)
IIASA	International Institute for Applied System Analysis
IN	Indicator (INPRO)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
INS	Innovative Nuclear Energy System (INPRO)
INSAG	International Nuclear Safety Advisory Group (IAEA)
IPCC	Intergovernmental Panel on Climate Change
ISED	Indicator for Sustainable Energy Development (IAEA)
J	Joule
J/s	Joule per second
kg	Kilogram (10^3 gram)
kJ	Kilojoule (10^3Joule)
km	Kilometer (10^3meter)
kW	Kilowatt (10^3Watt)
kW·h	Kilowatt-hour (10^3Watt-hour)
L	Litre
lb/MBtu	Pounds per mega (10^6) (British thermal unit)
LCA	Life cycle assessment
LCI	Life cycle inventory
LEU	Low enriched uranium
LFR	Lead cooled fast reactor
LMJ	Laser Mega-Joule (fusion energy facility in France)
LNG	Liquid natural gas
LOCA	Loss of coolant accident
LWR	Light water reactor
m	Meter
mA	Milliamp

mbd	Million barrels per day
MBtu	Mega Btu (10^6 Btu)
McF	Million cubic feet
meV	Milli-electron volt
MeV	Mega-electron volt
MFA	Material flow assessment
MJ	Megajoule (10^6 Joule)
MJ/s	Megajoules per second
MJ/s_e	Megajoules per second, electrical
MNFC	Multilateral fuel cycle (INPRO)
mph	Miles per hour
MSR	Molten salt reactor
Mt	Million metric tonne
MtC	Million metric tonnes of carbon
MW	Megawatt (10^6 Watt)
NCS	National Case Study (INPRO)
NGO	Non-governmental organization
NII	Investment needed for national infrastructure (INPRO)
NM	Nuclear material
NPP	Nuclear power plant
NPV	Net present value
NPT	Non-Proliferation Treaty
NRC	Nuclear Regulatory Commission (USA)
O&M	Operation and maintenance
P&T	Partitioning and transmutation
PHWR	Pressurized heavy water reactor
PIRT	Phenomena identification and ranking table
ppm	Parts per million
PR	Proliferation resistance (INPRO)
PRIS	Power Reactor Information System (IAEA)
ps	Picosecond (10^{-12} second)
PSA	Probabilistic safety analysis
psi	Pound per square inch
PUREX	Plutonium uranium extraction process
PWR	Pressurized water reactor
RBI	Relative Benefit Index (INPRO)

RBMK	Graphite moderated fuel channel reactor
RD&D	Research, development and demonstration
RG	Reactor grade
ROW	SRES region of rest of the world (beside OECD-90, Asia and REF)
s	Second
S/cm	Siemens/cm (unit of ionic conductivity)
SCWR	Supercritical water cooled reactor
SFR	Sodium cooled fast reactor
tC	Tonne of carbon
TcF	Trillion cubic feet
TE	Thermoelectric
TW	Terawatt (10^{12} Watts)
TW(e)	Terawatt electric
TW·h	Terawatt-hour
UREX	Simplified version of PUREX
UV	Ultraviolet
VHTR	Very high temperature (gas cooled) reactor
W	Watt

CONTRIBUTORS TO DRAFTING AND REVIEW

Busurin, Y.	International Atomic Energy Agency
Cherepnin, Y.	NIKIET, Russian Federation
Chun, K.S.	KAERI, Republic of Korea
Depisch, F.	International Atomic Energy Agency
Dyck, G.	AECL, Canada
Fukuda, K.	International Atomic Energy Agency
Ganguly, C.	International Atomic Energy Agency
Greneche, D.	Cogema, France
Uchiyama, G.	JNC, Japan
Weidinger, H.	Siemens KWU, Germany