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Assessment of Nuclear Energy Systems based on a Closed Nuclear Fuel Cycle with Fast Reactors

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

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ASSESSMENT OF
NUCLEAR ENERGY SYSTEMS BASED ON A CLOSED NUCLEAR FUEL CYCLE WITH FAST REACTORS

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AND FUEL CYCLES (INPRO)

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FOREWORD

A Joint Study was started in 2005 and completed in 2007 within the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). Canada, China, France, India, Japan, the Republic of Korea, the Russian Federation, and Ukraine participated in this study. The objectives were to assess a nuclear energy system based on a closed fuel cycle (CNFC) with fast reactors (FR) regarding its sustainability, determine milestones for the nuclear energy system deployment, and establish frameworks for, and areas of, collaborative R&D work. The assessment was carried out in accordance with requirements of INPRO methodology and guiding documents of the Joint Study developed and approved by the participating parties (Canada and Ukraine participated in the discussions during the Joint Study but did not contribute to the assessments themselves).

The Joint Study was implemented in steps. In its first step, nominated experts in course of extensive discussions analyzed the country/region/world context data, discussed national and global scenarios of introduction of the INS CNFC-FR, identified technologies suitable for the INS, and arrived at a broad definition of a common INS CNFC-FR. In the second step, the participants of the study examined characteristics of INS CNFC-FR for compliance with criteria of sustainability developed in the INPRO methodology in the domain of economics, safety, environment, waste management, proliferation resistance, physical protection and infrastructure. The results of the study were submitted to and endorsed by the INPRO Steering Committee meetings held in Vienna 2005–2007.

The authors of the report highly appreciate the valuable comments provided by delegates of INPRO Steering Committee meetings as well as the advice and assistance of the other experts.

Due to the length of the Joint Study report a summary of the results was produced, which was published as a hard copy. The full text of the Joint Study report is available on the CD-ROM attached this book.

The IAEA officer responsible for this publication was R. Beatty of the Division of Nuclear Power, Department of Nuclear Energy.

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Summary

The Joint Study was started in 2005 and completed in 2007. Canada, China, France, India, Japan, the Republic of Korea, the Russian Federation, and Ukraine participated. The objectives were to:

- assess the potential of a nuclear energy system based on a closed fuel cycle (CNFC) with fast reactors (FR) regarding its sustainability using the INPRO methodology;
- determine milestones for the CNFC-FR system deployment; and
- establish frameworks for, and areas of collaborative R&D work.

It was agreed to use for the assessment as a reference system a commercial CNFC–FR system, deployable in the near term (20 to 30 years) based on proven technologies, such as sodium coolant, mixed oxide (MOX) pellet fuel, and advanced aqueous reprocessing technology.

Results of the nuclear energy system assessment using the INPRO methodology

The INPRO methodology requires an assessment in seven areas to confirm the sustainability of a nuclear energy system: economics, infrastructure, proliferation resistance, physical protection[1], environment (impact by stressors and availability of resources), waste management and safety. The main results of the multidimensional assessment of the CNFC-FR system could be summarized as follows:

- *Availability of resources*: Recycling of plutonium (together with uranium) in spent fuel of CNFC-FR systems leads to practically inexhaustible resources of fissile material (and fertile material), i.e. such a system might de facto be considered as a renewable energy source. Globally, there is sufficient spent fuel available for reprocessing Pu to be used as fuel for FR. However, in some countries with an expected high growth rate of their national nuclear power program lack of spent fuel as a resource of Pu may impede an optimal deployment of CNFC-FR systems. Thus, the Joint Study concluded that a CNFC-FR system is well suited for and might require a regional or multilateral approach as no individual country participating in the Joint Study reflects the full set of factors that favour development and deployment of such a system. Examples of important favourable factors are predicted high growth of energy demand and large resources of Pu available in spent fuel.
- *Impact of stressors*: CNFC-FR systems avoiding mining/enrichment steps in their fuel cycle show a significantly reduced environmental impact caused by a much lower release of non radioactive elements compared to current licensed thermal reactor systems. Additionally, the radiation dose of CNFC-FR systems on the public is demonstrated to be far below regulatory limits.
- *Waste management:* The CNFC–FR system meets all INPRO requirements of an effective and efficient nuclear waste management. By recycling of specific (heat producing and long lived) nuclear fission products and minor actinides in addition to

[1] The area of physical protection of the INPRO methodology was not covered in the Joint Study as the description of the assessment method for this area was not yet available during the performance of the Joint Study.

plutonium (together with uranium), the CNFC–FR system has the potential to significantly reduce the heat load, mass/volume and radiotoxicity of high level waste to be deposited. The reduction of heat load enables to store more waste per volume of rock, and the removal of actinides and specific fission products from the waste decreases the time required to manage nuclear high level waste from a geological time scale (several 100000 years) to a civilization time scale (several 100 years). However, in comparison to a once through fuel cycle (OTFC) reprocessing of spent fuel in a CNFC produces several additional secondary nuclear waste forms (e.g., losses in the processes) most of them needing geological disposal.

- *Safety:* Safety characteristics of the CNFC-FR system meet the current safety standards. A comparison of a CNFC-FR with a thermal reactor system showed that disadvantages of the fast neutron system were compensated by its inherent safety features and additional engineered safety measures. A probabilistic analysis performed for the Russian BN-800 fast reactor design confirmed that its innovative design features lead to a significant reduced risk of severe accidents, thus relieving the need for relocation or evacuation measures outside the plant site.

- *Proliferation resistance:* CNFC–FR systems show several features that result in comparable or higher proliferation resistance compared to thermal reactors with a once-through fuel cycle (OTFC). The higher proliferation resistance of CNFC-FR is justified by eliminating enrichment of uranium and avoiding the accumulation of Pu in spent fuel (“plutonium mines”) in an OTFC, excluding Pu separation in advanced reprocessing technologies, and by the possibility to produce fresh fuel with a high radiation level and to reduce fuel transportation via collocation of FR and fuel cycle facilities applying pyro-processing technology. The use of a higher fissile content in the FR fuel results in a decrease of proliferation resistance in comparison to thermal reactor systems.

- *Infrastructure:* The INPRO basic principle in the area of infrastructure asks for the availability of regional and international arrangements to limit the necessary effort to establish the necessary infrastructure for a nuclear energy system. As stated above in the area of availability of resources, the Joint Study concluded, a CNFC–FR system is well suited for and might require such new regional or international arrangements as it is capable of converting spent fuel of all reactors into a valuable energy resource, thereby offering the opportunity expanding fuel cycle front end and backend services on a multinational basis to technology holder as well as to technology user countries. Looking at the national legal infrastructure of the countries participating in the Joint Study it was found that the legal frame work needed to operate a nuclear energy system is well established and is deemed sufficient to cover also future CNFC-FR systems. However, regional or international approaches might require new international legal infrastructure. The industrial infrastructure and human resources to design, manufacture, construct and operate a CNFC-FR system are available in most countries participating in the Joint Study.

- *Economics:* The designs of currently operating fast reactors with a closed fuel cycle are not completely economically competitive against thermal reactor systems or fossil power systems due to high capital costs. However, the necessary modifications of the design, such as simplifying the design, increasing the fuel burnup, constructing small series are integrated into the development programs of all Joint Study participants. These modifications will make electricity costs produced by CNFC-FR systems comparable to those of thermal reactor and fossil fuelled power plants.

Milestones of instalment of CNFC-FR systems

China plans to increase its nuclear capacity significantly from about 7 GW_e to 60 GW_e by 2030 installing mainly PWR of different sizes. Starting around 2020, fast breeder reactors are planned to be added and these will become the dominant type of nuclear reactors by the end of the 21st century. The cores of the fast reactors are designed with moderate breeding rates ~1.1 to 1.2 at the beginning and could be replaced later by core designs with high breeding rates such as ~ 1.5.

In *France*, for the purpose of the study, nuclear power capacity of about 63 GW_e is assumed to remain constant throughout the 21st century. The existing fleet of PWR is to be replaced by about 2030 with EPR (Generation III+) reactors. Thereafter fast reactors to be operated as Pu burners (with a conversion rate about 1) should become the dominant option of nuclear energy supply.

India plans to build primarily PHWR and PWR to satisfy the predicted large increase of capacity of nuclear power from 3 GW_e to about 30 GW_e by 2020. Starting in 2010 a fleet of fast breeder reactors with – similar to China – a moderate breeding ratio of ~ 1.1 to 1.2 at the beginning is planned to be installed, and later changed to core designs with a high breeding ratio of ~ 1.5. Finally, advanced PHWR – using an advanced thorium fuel cycle – are foreseen to be installed leading to a total nuclear capacity of 275 GW_e by 2050.

Japan – similar to France – expects a moderate increase of nuclear generation capacity from 47 GW_e to 60 GW_e by 2030. Thereafter it is foreseen to keep the nuclear generation capacity constant till the end of the 21st century. Starting around 2050 fast breeder reactors (with a low breeding ratio of ~ 1.03 to 1.10) could gradually replace LWR.

The *Republic of Korea* investigates several options for satisfying the predicted increased capacity of nuclear power from about 17 GW_e to 27 GW_e by 2015. One option foresees the instalment of additional water cooled reactors only, and the two other options include the instalment of fast reactors to be operated as Pu burners (with a conversion rate about 1) after about 2030.

In the *Russian Federation* the nuclear capacity should increase significantly from about 22 GW_e to 81 GW_e by 2050. The existing fleet of VVER is to be replaced by Generation III reactors of type AES2006 and around 2030 fast breeder reactors are foreseen to replace gradually the thermal reactors. Breeding ratios are expected to be within a wide range from about 1.0 to 1.6 at different stages of the nuclear power program.

Looking at the development programs of a CNFC-FR system in the countries participating in the Joint Study it is to be noted that:

- All countries developing fast reactor technology selected a stepwise introduction of the technology, starting with a small experimental reactor (< 50 MW_{th}) to test the feasibility of the concept, then installing a prototype (several 100 MW_{th}) to confirm all technical issues are resolved, thereafter constructing a first commercial size reactor (several 1000 MW_{th}) to proof its competitiveness, and finally install a series of commercial reactors by 2020 to 2050. A similar stepwise approach is applied for the development of the associated fuel cycle technology.

- The time schedule of instalment of the CNFC-FR system strongly depends on the global as well as on the national development of nuclear power. The higher the growth rate of nuclear power capacity either assumed globally or planned in the country, the earlier the installation of a CNFC-FR system is required to assure the availability of cheap fissile material. Differences between countries in the installation schedule are mainly caused by different predicted growth rate of the national nuclear power program.
- Countries with a large nuclear power program established for a long time and with moderate or no planned increase of their nuclear power capacity, i.e. France, Japan, or Korea, expect to accumulate enough Pu in spent fuel needed for a fast breeder program, and are therefore focusing on the development of core designs of FR with low to moderate breeding rates, e.g., Pu burners.
- Countries like China, India and Russia with rather limited contribution of nuclear power to their total energy supply but with a planned large and rapid increase of their nuclear power capacity expect not to accumulate enough Pu in spent fuel for their planned FR program, and therefore aim initially already at core designs with moderate breeding ratios and consider in the long term core designs with breeding rates as high as possible to avoid a shortage of fissile material.

R&D defined in the Joint Study

The *Joint Study* (JS) concluded that a comprehensive program of R&D is absolutely essential in a variety of areas (especially, for economics and safety) with an inter-disciplinary approach and international collaborations wherever possible to make a CNFC-FR system a viable alternative to conventional sources of power.

As capital costs of currently operating (sodium cooled) FR were 40 % up to three times higher than capital costs of thermal reactors, several possibilities for reduction of capital costs were presented.

For the improvement of FR safety, R&D is needed to develop efficient and cost-effective shielding materials such as boride/rare earth combinations, and achieve (radiation) source reduction by adequate measures such as use of materials which do not get activated.

In the Joint Study the following INPRO collaborative projects related to fast reactors have been proposed and are currently underway:

- A Global Architecture of nuclear energy systems based on thermal and fast reactors including a closed fuel cycle (called **GAINS**).
- Integrated Approach for the design of safety grade decay heat removal system for liquid metal cooled reactor (called **DHR**);
- Assessment of advanced and innovative nuclear fuel cycles within large scale nuclear energy system based on CNFC concept to satisfy principles of sustainability in the 21st century (called **FINITE**);
- Investigation of technological challenges related to the removal of heat by liquid metal and molten salt coolants from reactor cores operating at high temperatures (called **COOL**);

Feedback from the Joint Study on the INPRO methodology

Besides detailed proposals how to improve the INPRO methodology in specific areas several general proposals have been made to improve the INPRO methodology as set out below.

The INPRO methodology should be extended to enable a clearer distinction (discrimination) between different options of nuclear energy system components under development.

An approach should be developed how to treat different level of uncertainty associated with stages of development.

In particular for the INPRO area of environment and proliferation resistance a need for further development of the assessment approach was expressed.

CHAPTER 1
INTRODUCTION

This project, called the Joint Study, was initiated by the Russian Federation in 2004, started in 2005 and completed in 2007. It was part of Phase-1 (2001 to June 2006) of the International Project for innovative reactors and fuel cycle technologies (INPRO) and was implemented by Canada, China, France, India, Japan, Republic of Korea, Russian Federation, and Ukraine. Based on a decision of its Steering Committee in June 2006 the INPRO project is currently in Phase-2.

The main objectives of the Joint Study were to:

- assess the potential of a nuclear energy system consisting of a closed nuclear fuel cycle and fast reactors (CNFC-FR) for satisfying the criteria of sustainability as defined in the INPRO methodology;
- determine milestones for deployment of CNFC-FR systems; and
- identify areas for collaborative R&D work.

The assessment was carried out using the INPRO methodology as documented in the IAEA reports IAEA-TECDOC-1434 [1] and IAEA-TECDOC-1575 [2].

The Joint Study was implemented in steps. In the first step, experts from the participating countries discussed national and global energy scenarios for the introduction of a CNFC-FR system, identified technologies suitable for such a CNFC-FR system, and defined a common CNFC-FR nuclear energy system called a “reference system” to be used for the joint evaluation.

In the second step, the participants of the Joint Study examined characteristics of the “reference system” for compliance with criteria of sustainability developed in the INPRO methodology in the areas of economics, infrastructure, waste management, environment, proliferation resistance, and safety. The interim results (Ref. [3]) of the Joint Study were submitted to and endorsed by the INPRO Steering Committee in the course of 2005-2007. Final results of the Joint Study are summed up in this report.

Chapter 2 of the report provides an overview of the national nuclear energy policies of the countries participating in the Joint Study including possible scenarios of deployment of fast reactors with a closed fuel cycle (CNFC-FR system).

Chapter 3 presents a comparison of national energy policies and discusses differences and commonalities in the national approaches to develop and install a CNFC-FR system.

Chapter 4 lays out the results of the assessment of the CNFC-FR system applying the INPRO methodology in the area of economics, infrastructure, waste management, proliferation resistance, environment and safety.

Chapter 5 provides a short description on some important R&D issues necessary to develop a commercial version of a CNFC-FR system.

Annex A presents feed back from the assessment study on the INPRO methodology.

CHAPTER 2
NATIONAL NUCLEAR ENERGY SCENARIOS

In this chapter a short overview of the national energy scenario of the countries[3] participating in the Joint Study is presented. The planned role of nuclear power and especially of fast reactor systems with a closed fuel cycle is discussed in more detail.

2.1. National nuclear energy scenario of China

According to the goal of the national development, China's gross domestic product (GDP) should double from 2000 to 2020 resulting in a corresponding growth of energy demand. To satisfy the need for electricity by 2020 the installed capacity of power plants should increase from 400 GW_e to a range of 960 to 1000 GW_e. More than half of this increase is to be filled by coal fired plants and about a third is to be contributed by hydro plants. The remaining power should be supplied primarily by nuclear power, i.e. the nuclear capacity should be increased from the current 9 GW_e to about 40 GW_e by 2020. In 2009, China is expected to adjust its goal of nuclear power development further to 70 GW_e to be installed by 2020.

After about 2020, it is predicted that the further expansion of the use of coal will be limited by the environmental burden to a maximum capacity of ~900 GW_e by 2050, and there will be no big room for further expansion of hydro-power after its capacity reaches a maximum of ~300 GW_e. Other renewable resources (such as wind and solar energy) will be fully encouraged to develop and will play an increasing role, especially in some remote or coastal areas. But it will be unlikely that renewable energy resources could replace fossil resources in a large scale in the foreseeable future. Gas could contribute about 100 GW_e by 2050. The major way to meet the national energy demand after 2020 (about 1600 GW_e by 2050) is thus to further increase the share of nuclear energy thereby making an important contribution to both energy security and environmental safety.

By the middle of the 21st century, the capacity of nuclear power in China should reach ~200 GW_e or even higher with a share up to 15 % of the total electrical power capacity. Nuclear power will then be one of the three major energy resources in China together with "cleaned" coal and hydro-power.

To supply such a large increase of nuclear capacity it is judged impractical to use only thermal reactors. Therefore, as shown in Figure 2.1, only until about 2020 power will be primarily generated by thermal reactors reaching a maximum capacity of ~150 GW_e (a value kept probably constant thereafter). Beginning in 2020, commercial fast breeder reactors together with the associated nuclear fuel cycle technologies are planned to be installed and become the primary source of nuclear electricity by about 2100.

Fast breeder reactors will play a role in China to maximize the energy potential of the national uranium resources, and fast burner reactors could transmute the minor actinides partitioned from spent nuclear fuel to minimize the volume and toxicity of high level waste requiring geological disposal.

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[3] Canada and Ukraine participated in the discussions during the Joint Study but did not contribute to the assessments themselves.

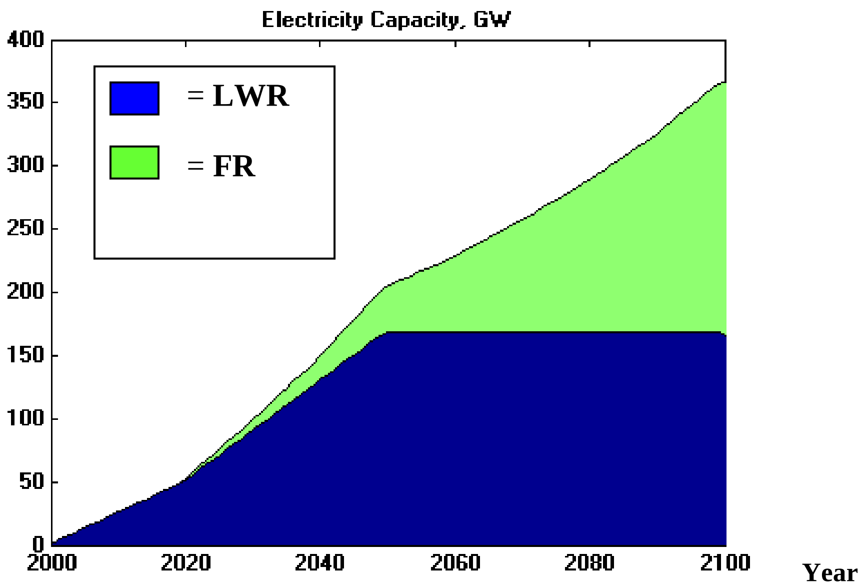

Figure 2.1. Simulated nuclear energy development scenario in China (optimistic case[4]).

China's fast breeder development program is carried out in steps. The China Experimental Fast Reactor (CEFR, 25 MW_e) is under construction and should reach first criticality in 2009. The China Prototype Fast Reactor (CPFR, 600 MW_e) is planned to start operation by 2020 and the China Demonstration Fast Reactor (CDFR, 1000 to 1500 MW_e) by 2035. The first cores of fast reactors will use high enriched uranium or MOX fuel. Later this fuel could be replaced by metal fuel to facilitate high breeding rates.

The facilities of the nuclear fuel cycle are also to be developed step by step. China's spent fuel reprocessing pilot plant with a capacity of 50~100 t_{HM}/a is under construction and will be put into trial operation by 2009. A commercial reprocessing plant is under consideration and expects to be built by around 2020. The reprocessing could be done using an advanced aqueous process with increased proliferation resistance by non separation of uranium and plutonium. Separation of minor actinides in spent fuel is also being considered.

The development of MOX fuel fabrication is now at an early stage. A 500 kg/a MOX fuel fabrication experimental line is under construction and will be put into operation by 2010. For the CEFR it will produce test fuel firstly and later the driver fuel. The test fuel for CPFR will also be produced by this experimental line. A commercial MOX fuel fabrication plant with a capacity of 40～100t/a is under consideration with the aim of commissioning it by about 2020.

[4] In the Joint Study also a moderate case was considered with an installed maximum nuclear capacity of about 100 GW_e of LWR and about 70 GW_e of FR by 2100.

2.2. National nuclear energy scenario of France

Early in 2003 France's first national energy debate was announced. The debate was to prepare the way to define the energy mix for the next 30 years in the context of sustainable development at a European and at a global level. The role of nuclear power was central to this, along with specific decisions concerning the European Pressurised Water Reactor (EPR), and defining the role of renewable energies in the production of electricity, in thermal applications and transport.

Currently, nuclear power amounts to about 80 % of total installed electrical capacity in France. For the purpose of the Joint Study the contribution of nuclear power to the national energy supply was assumed to be constant during the 21st century with a capacity of about 63 GW_e. Starting by 2020, the existing fleet of reactors and associated fuel cycle technologies are to be replaced gradually by a new generation of light water reactors, the EPR, and finally by Generation IV fast neutron systems by about 2035 (as shown in Figure 2.2).

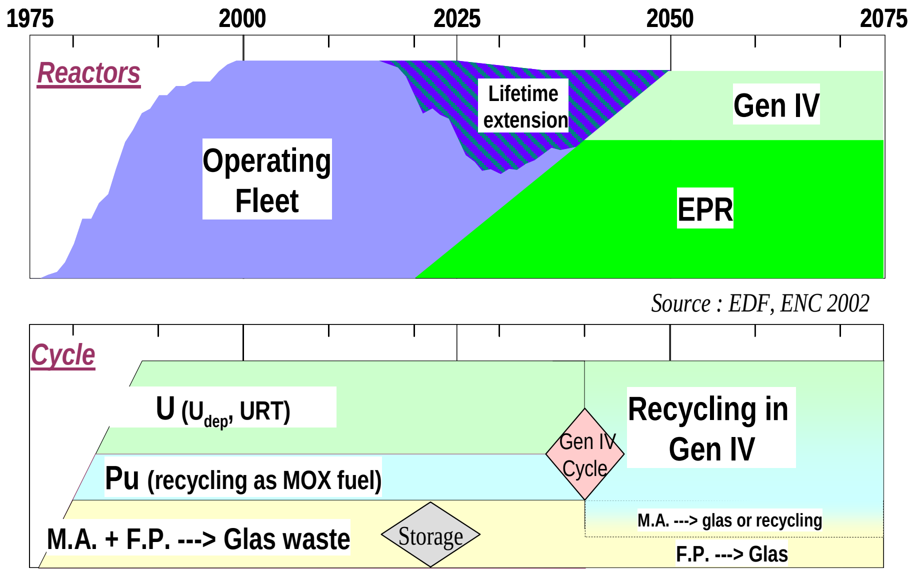

Figure 2.2. Evolution of the French nuclear reactors and fuel cycle technologies.
(Gen IV = Generation IV reactors, U_{dep} = depleted uranium, U_{RT} = reprocessed uranium, M.A. = minor actinides, F.P. = fission products)

The R&D strategy for the development of future nuclear energy systems is focused on three components:

- The development of fast reactors with a closed fuel cycle;
- The development of the very high temperature reactor for nuclear hydrogen production by the water splitting process and for supply of very high temperature process heat to the industry; and
- Innovations of fuel and reactor systems for light water cooled reactors.

France started its fast reactor development program [4] with the installation of the RAPSODIE (40 MW_{th}) reactor that operated from 1967 to 1983. In 1973 the prototype fast reactor PHENIX (560 MW_{th}, 250 MW_e) started up and is still in operation today. From 1985 to 1998 a commercial fast breeder reactor SUPERPHENIX (3000 MW_{th}, 1240 MW_e) was in

operation. The design of the European fast reactor (EFR) was completed in 1993 but the project was stopped in 1998.

The current development of fast reactors with closed fuel cycle includes two tracks:

- Sodium cooled fast reactors using MOX fuel with low breeding ratios; a prototype should start up by 2020; and
- Gas cooled fast reactors with the objective to develop an experimental reactor in the European framework.

The primary goals of the French fast reactor program are to increase efficiency of use of nuclear resources and to reduce the radiotoxicity of the high level waste to be put in final storage.

The reactor development tracks are associated with the development of new processes for spent fuel treatment and recycling. The new processes focus on the development of advanced aqueous reprocessing techniques that guarantee the non separation of uranium and plutonium in the process, and simultaneously enable the partitioning of minor actinides and specific fission products (that are long lived and produce high radiation) to be recycled (transmuted)in the reactor system.

2.3. National nuclear energy scenario of India

A survey by the Department of Atomic Energy forecasts electricity growth rate of 6.3 % per year till 2020 and about 4.0 % /y from 2020 to 2050. Total installed electrical capacity should reach ~1300 GW_e by 2050, 800 GW_e supplied from coal fired plants, 150 GW_e from hydro plants, 100 GW_e from non conventional plants (renewables) and 270 GW_e from nuclear power plants. If India relies only on domestic resources this significant increase in nuclear capacity can be achieved only by choosing a closed nuclear fuel cycle with a highly efficient use of domestic resources.

India has a vision of a three stage nuclear power program as shown in Figure 2.3.

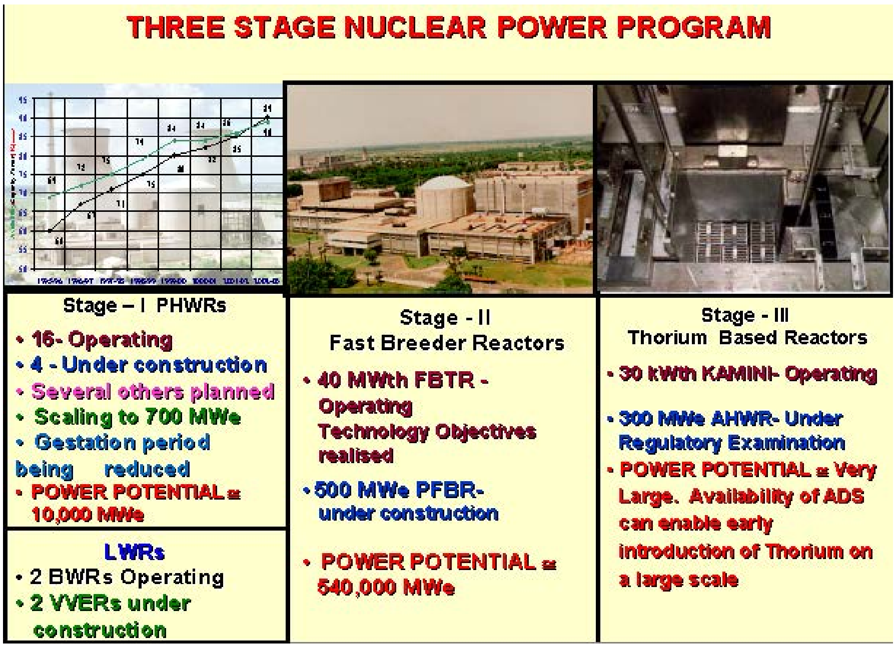

Figure 2.3. A three stage program of nuclear power development in India.

In the first stage, Pressurized Heavy Water Reactors (PHWR) are employed which will use natural uranium as fuel. PHWR design is already commercially proven in India and 16 units are in operation. Liquid sodium cooled fast breeder reactors (FBR) with high breeding rates form the second stage of the nuclear power program. Plutonium (Pu) generated in PHWRs would be used as the fissile material within a closed fuel cycle program in these FBR. Subsequently, thorium resources will be utilized in the third stage reactors, i.e. in Advanced Heavy Water Reactors with fuel containing Pu and U^{233} as fissile material produced in FBR.

India started its fast reactor development program by starting up an experimental fast breeder test reactor FBTR (40 MW_{th}) in 1985 that is in operation till today. A unique feature of this reactor is its plutonium carbide fuel.

Based on experience gained in the FBTR and in other international fast breeder programs a 500 MW_e prototype fast breeder reactor is currently under construction to be loaded with MOX fuel. It is planned to take up a comprehensive development program of advanced oxide and U-Pu-Zr metal alloy fuels for FBRs and associated fuel cycle technologies as the chosen option of shifting to a high breeding path by 2020.

After 2025 a series of 1000 MW_e commercial fast breeder reactors with a matching fuel cycle is planned.

2.4. National nuclear energy scenario of Japan

Japanese total energy consumption grew strongly in the past till 2005, but within the next 30 years it is expected to peak by 2021 and then could decrease because of population decline and transformation of economic and social structure. On the other hand, electricity demand is foreseen to continue to increase steadily from 940 TWh in 2000 to 1220 TWh by 2030. Though the ratio of oil in total primary energy supply should fall gradually from 47 % in 2000 to 38 % by 2030, it will remain the important energy resource for Japanese energy supply by 2030. It is anticipated that the share of natural gas in total power generation will rise to 30 % by 2030.

Japan imports most of its energy resources (approximately 96 %). To improve this situation, Japan has developed nuclear power for the last fifty years [5] based on the principle of peaceful use, and now (June 2006) 55 nuclear power plants are in commercial operation with a total installed capacity of about 50 GW_e; two nuclear power plants are under construction and eleven are under planning. By 2030 the nuclear power generation capacity is expected to increase to 58 GW_e and is assumed to remain constant thereafter.

The nuclear energy policy of Japan is characterized by the following objectives:

- Current contribution of nuclear power generation to the national electricity generation should be kept constant or increase from 30 % to 40 % by 2030;
- Advanced light water reactor designs should be developed for replacement of existing NPP beginning around 2030;
- Commercial operation of fast reactors (FR) should be achieved close to 2050 on the premise of meeting the necessary conditions; and
- Traditional national nuclear policy should be kept, i.e. spent fuels are reprocessed and recovered plutonium and uranium are used effectively.

Japan has promoted the development of a closed nuclear fuel cycle to enhance the efficient use of uranium resources and to reduce high-level radioactive wastes as a national policy. A

1050 10^3 SWU/year enrichment plant and a low-level radioactive waste disposal facility are in operation. Commercial operation of the Rokkasho reprocessing plant with annual throughput of 800 t_{HM} is scheduled to begin in 2009. The construction of a mixed-oxide (MOX) fuel fabrication plant is also in progress at the Rokkasho site. Plutonium extracted from the reprocessing of spent fuel will be recycled into light water reactors as MOX fuel. The legal framework of the disposal of high-level radioactive wastes was promulgated in 2000. Potential sites are now being surveyed in accordance with the law, and construction and operation of facilities are planned to commence by the late 2030s.

Japan's vision of the transition from a thermal to a fast reactor cycle is shown in the following Figure 2.4.

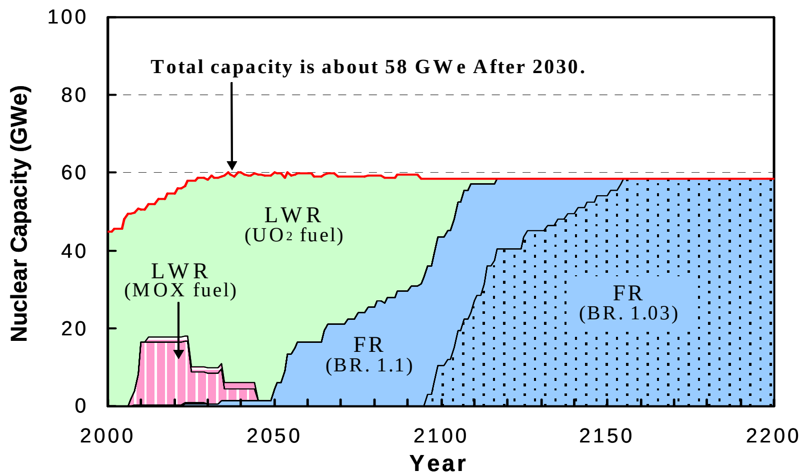

Figure 2.4. Transition scenario from LWR cycle to FR cycle in Japan.

The driving force behind the Japanese fast reactor program is the goal to further reduce the need to import natural uranium, and to minimize volume, mass and radiotoxicity of high level radioactive waste (see Figures 2.5 and 2.6) by developing advanced reprocessing technologies such as advanced aqueous reprocessing or pyro-reprocessing.

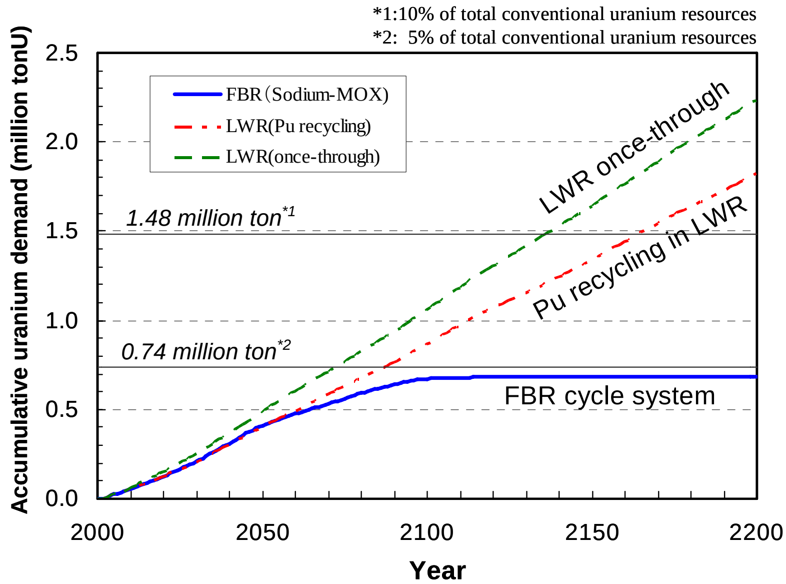

Figure 2.5. Cumulative natural uranium demand in Japan.

Figure 2.5 above demonstrates the possible reduction of imported natural uranium by introduction of a closed fuel cycle and a fast reactor system in Japan.

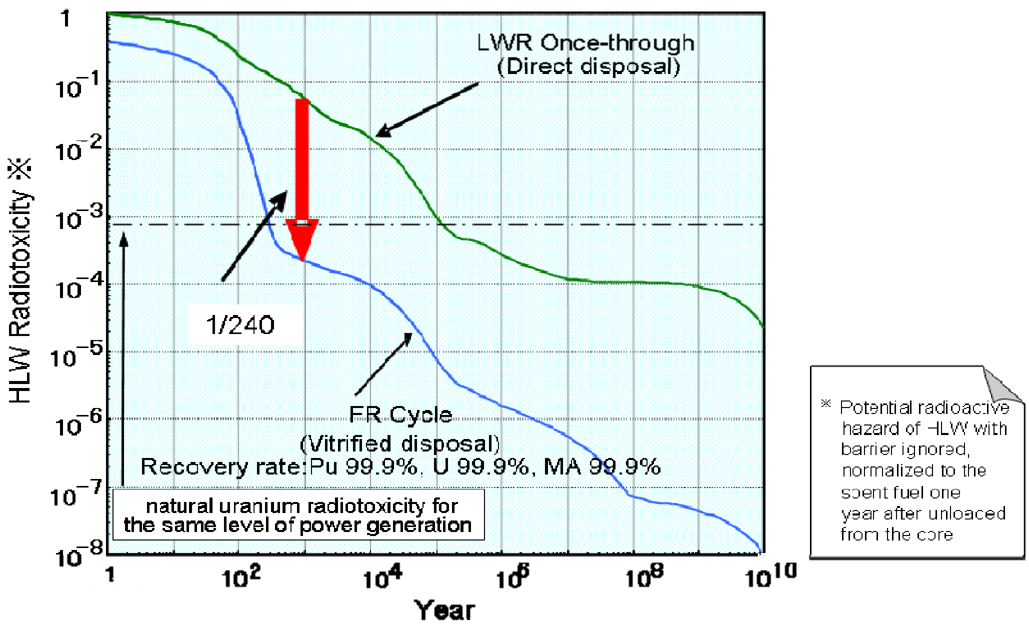

Figure 2.6. Reduction of radiotoxicity of high level waste in Japan.

Figure 2.6 above illustrates the potential of a fast reactor system to reduce the radiotoxicity of high level waste by recycling of all actinides.

Japan's fast reactor development program is following a step-by-step approach starting with an experimental fast reactor JOYO that started up in 1977. This reactor has been in operation successfully with different core designs, i.e. starting with Mk-I (50 MW_{th}), then Mk-II (100 MW_{th}), and most recently Mk-III (140 MW_{th}).

The prototype sodium cooled fast reactor MONJU (280 MW_e, 714 MW_{th}) was connected to the grid in August 1995. However, in December 1995 a sodium leak occurred in the secondary heat removal system and the plant has remained shutdown after that incident. Several modifications of the plant's systems were performed in the meantime and it is expected to start up again in 2009.

To establish the design of a commercial fast reactor a feasibility study has been performed in Japan from 2000 to 2005. Based on the results of this study a follow up program called Fast Reactor Cycle Technology Development (FaCT) project was initiated. Within this project a demonstration fast reactor is being developed that should be operating by around 2025 and will provide experience for the commercial fast reactor to start up by 2050.

2.5. National nuclear energy scenario of the Republic of Korea

Due to economic growth and the desire for a better quality of life there is an ever increasing demand for electricity in Korea. The estimated average annual growth rate of the generation capacity is estimated to be 2.5 % during the period from 2006 to 2017. The availability of conventional energy resources such as hydro or coal is extremely limited in Korea and therefore nuclear power plants play a very important role in achieving energy self-reliance and in stabilizing the electricity price.

In the year 2005, nuclear power plants occupied 28 % (17.7 GW_e) of the total installed capacity (about 62 GW_e), but generated as much as 40 % of the total electricity. By 2017 nuclear power will represent about 33 % of installed electrical capacity and by 2100 installed nuclear capacity is projected to reach a value of about 60 GW_e.

Recognizing the capabilities of fast reactors systems to improve the utilization of uranium resources and significantly reduce high level radioactive waste a Korean development program for fast reactors was started in 1992.

The design of a fast reactor – sodium cooled, pool type, metal fuel – called KALIMER with different sizes (150 MW_e and 600 MW_e) was developed until 2007. The ongoing program aims at developing a sodium cooled fast reactor that is fully consistent with the Generation-IV requirements of Generation IV International Forum. The KALIMER-600 design served as a starting point for this program.

Together with the reactor design also the corresponding advanced fuel cycle technologies are developed such as pyroprocessing of spent nuclear fuel.

A study was performed to demonstrate the advantages of installing fast reactors as burners of all transuranic elements (Pu and MA). Three different scenarios were compared:

- Case-1: Once-through strategy; spent fuel from PWR and CANDU is directly disposed.
- Case-2: Partial deployment of KALIMER reactors; PWR spent fuel reprocessing begins in 2020; starting in 2030 PWR will be gradually replaced by KALIMER reactors; KALIMER reactors will reach a 21 % fraction of total nuclear energy capacity in 2100.
- Case-3: Full deployment of KALIMER reactors; PWR spent fuel reprocessing begins in 2020. KALIMER reactors are deployed gradually instead of PWR from 2030 on and will reach a 76 % fraction of total nuclear energy capacity in 2100. All new nuclear plants will be KALIMER reactors after 2090.

The following Figure 2.7 shows the accumulation of spent fuel for the Case-1 scenario, i.e. a once through or an open fuel cycle strategy.

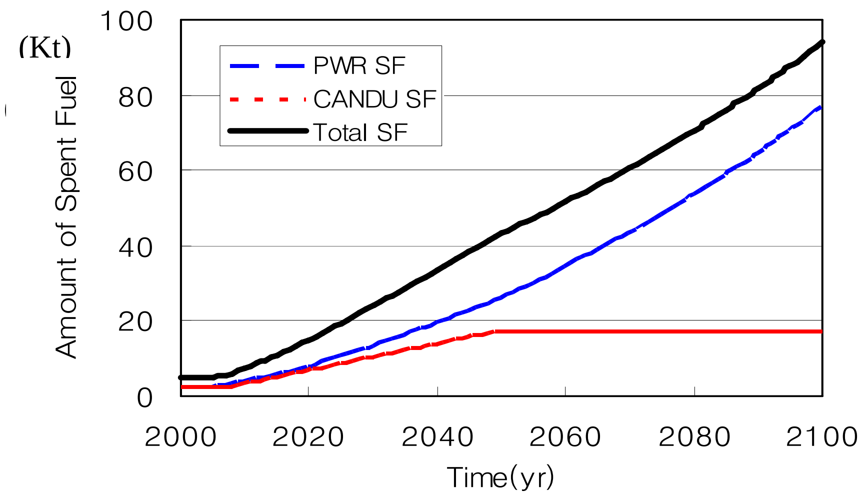

Figure 2.7. Accumulation of spent fuel in the Case-1 scenario in Korea. (SF=Spent Fuel)

Choosing the Case-2 or Case-3 scenario with introduction of fast reactors with a closed fuel cycle leads to significant reductions of spent fuel as shown for Case-3 in the following Figure 2.8.

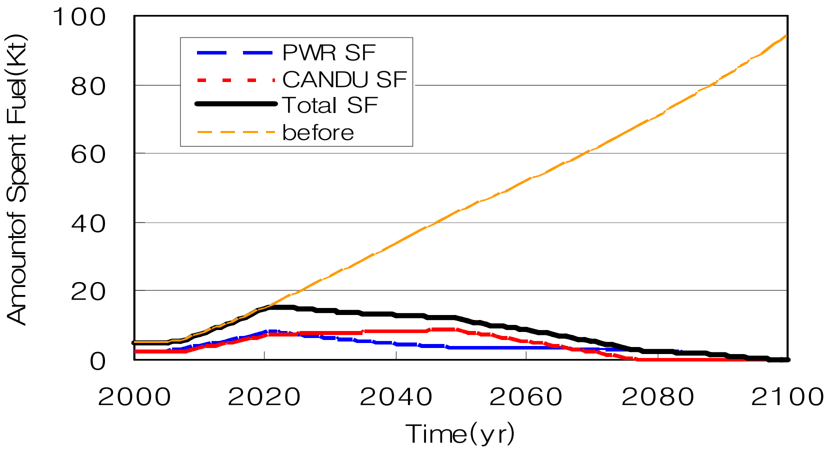

Figure 2.8. Accumulated spent fuel in the Case-3 scenario in Korea. (before=Case-1)

2.6. National nuclear energy scenario of the Russian Federation

In Russia, currently, about 50 % of the total electricity is generated by burning natural gas, while coal provides about 18 %, and the share of electricity generated by hydroelectric power plants is 16 %, equal to that of nuclear plants. The development plan of the energy sector foresees a doubling of the electricity production by 2020 with a goal to decrease the dependence on limited resources of gas.

To satisfy the growing demand for energy the share of nuclear power within the energy mix should be increased significantly within the next decades from currently about 16 % (23 $GW_{e)}$ to about 25 % (80 GW_e) by 2050. Analyzing different scenarios with different types of reactors and associated fuel cycles it was concluded [6] that the introduction of fast breeder reactors with a high breeding ratio (1.2 to ~1.6) and a closed fuel cycle both for thermal and fast reactors can reduce significantly the needed amount of natural uranium and enrichment. Thus, the vision of the development of the nuclear power program in Russia looks as shown in the following Figure 2.9. Two possible installation programs for fast reactors are foreseen, one with 1.2 GW_e added per year, the other one with 3.6 GW_e per year, beginning by about 2030.

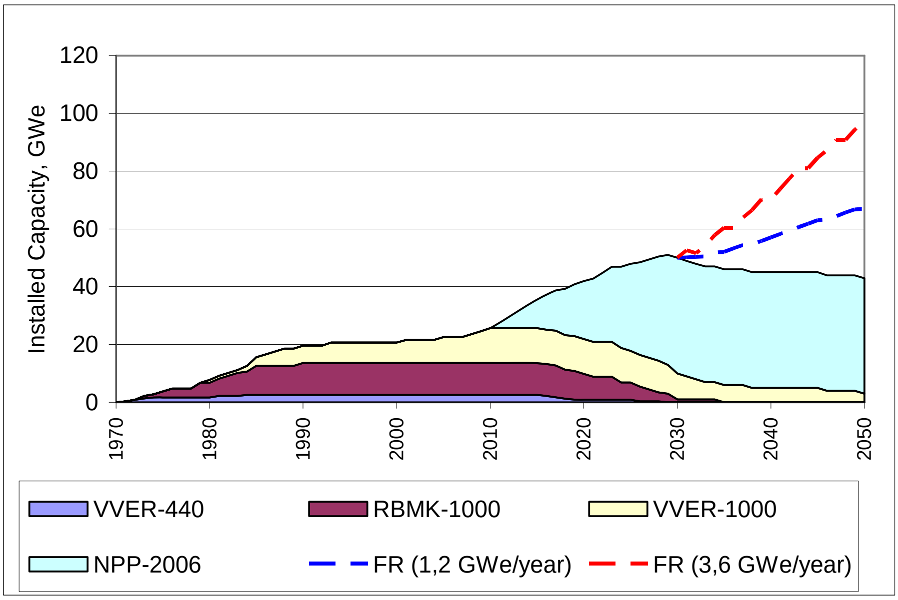

Figure 2.9. Russian nuclear power program (retrospective and forecast). (FR= fast reactor, NPP-2006= Generation III PWR)

The development of fast reactors in Russia started in 1954 at Obninsk with the mercury cooled test fast reactor type BR-1 that was upgraded to BR-2 (0.1 MW_{th}) in 1956 (shut down in 1957), followed by the sodium cooled fast reactor BR-5 (5 MW_{th}) in 1959 upgraded to the BR-10 (8 MW_{th}) in 1973 that operated till 2002. At Dimitrovgrad the fast test reactor BOR-60 (sodium cooled, MOX, fuel, 55 MW_{th}) started up in 1968 and is still operating. The sodium cooled prototype fast reactor BN-350 (UO_2 fuel, 150 MW_e[5], 750 MW_{th},) operated from 1973 to 1999 in Chevchenko producing electricity and heat for industrial applications and desalinated water.

The sodium cooled fast reactor BN-600 (UO_2 fuel, 600 MW_e, 1470 MW_{th}) started up at Beloyarsk in 1980 and is still in operation. The follow up design called BN-800 (MOX fuel, 800 MW_e, 2100 MW_{th}) is currently under construction (to be commissioned by 2012) and a BN-1600 and a BN-1800 are being designed.

A specific characteristic of the fuel cycle associated with fast reactors in Russia is the installation of on site fuel cycle facilities including non aqueous reprocessing, fuel rod manufacturing using vibrocompaction processes, and waste management facilities.

[5] BN350 produced heat for industrial applications including desalination in addition to electricity.

CHAPTER 3
COMPARISON OF NATIONAL ENERGY SCENARIOS

This chapter compares the national energy scenarios and the national approach to develop a fast reactor system of the countries participating in the Joint Study.

3.1. Comparison of primary energy resources and structures of power supply

The availability of domestic energy resources differs greatly in the countries that participated in the Joint Study (see Table 3.1). But, the energy policy of all the countries is focused on extending their resource base. Alternative energy sources like wind, geothermal, solar and biomass are recognized to provide opportunities as a local small-to-medium size power source, but they are deemed not to be capable of meeting increasing energy demands in a sufficient manner.

Table 3.1. Proven reserves of the main primary energy resources in the countries of the Joint Study (2006) [7]

	Oil (10^9 barrels)	**Natural gas** (10^{12} m^3)	**Coal** (10^6 tonnes)	**Hydropower** (10^{12} Watt-hours/year)	
				Gross theoretical capability	Technically exploitable capability
China	16	3	120000	6100	2500
France	0	0	15	270	100
India	6	1	93000	2600	660
Japan	0	0	360	720	140
Korea	0	0	80	50	26
Russia	80	48	160000	2300	1700
Canada	17	2	7000	2200	1000
Ukraine	0	1	34000	50	24

The countries participating in the Joint Study have a large scale power supply system in place. However, there are significant differences in power system structures among the countries as shown in Figure 3.1.

Using nuclear power as an energy source seems inevitable for countries having large-scale power systems and a lack of indigenous energy resources. However, even for the countries possessing substantial reserves of fossil fuels and at the same time having large-scale national power systems, the use of nuclear power is advantageous since this option provides an opportunity to enhance energy security and sustainability features of the internal energy sector while preserving its export potential. But differences in power structure and indigenous energy resources lead to different national strategies for nuclear power deployment.

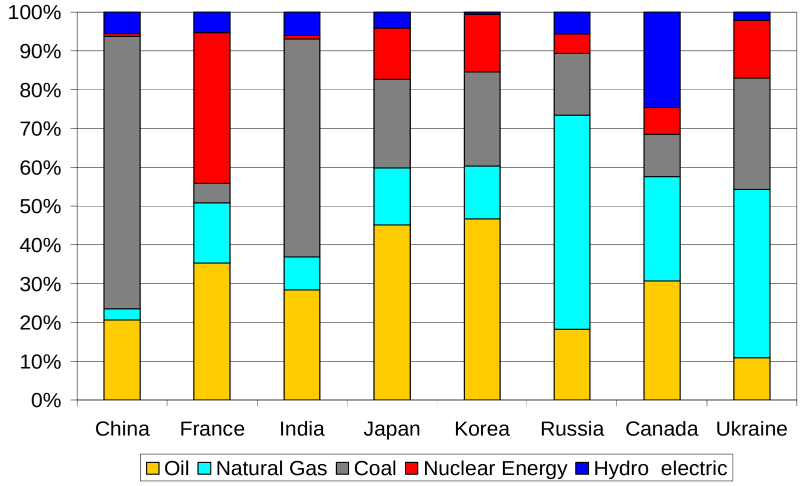

Figure 3.1. Distribution of primary energy consumption in the countries participating in the Joint Study [8].

Existing nuclear power systems play an important role in the energy supply of each of the countries participating in the Joint Study, even though the current share of nuclear in electricity generation varies significantly, from about 2 % in China up to nearly 80 % in France. Thermal reactors of various designs dominate. They operate mainly in base load mode and using a once-through fuel cycle and uranium oxide fuel. A summary of the reactor types in operation in the countries of the Joint Study is shown in Table 3.2.

Table 3.2. Gross capacities of nuclear power reactors in MW_e installed in the countries of Joint Study, (2007) [9]

	BWR	FBR	PHWR	PWR	RBMK
China	-	-	1400	7468	-
Canada	-	-	13817	-	-
France	-	250	-	66056	-
India	420	-	3995	-	-
Japan	30214	-	-	19366	-
Korea	-	-	2785	15339	-
Russia	-	600	-	11618	11000
Ukraine	-	-	-	13880	-

BWR= boiling water reactor; FBR=fast breeder reactor; PHWR=pressurized heavy water reactor; PWR=pressurized water reactor; RBMK=light water cooled graphite moderated fuel channel reactor

3.2. Comparison of national approaches for development of CNFC-FR

Looking at the national energy scenarios of the participating countries[5] in regard to development and deployment of fast reactors and corresponding closed fuel cycles, as described in Chapter 2, several commonalities and also a few significant differences can be noted.

Table 3.3 provides a summary of the national approaches towards a nuclear energy system consisting of fast reactors with a closed fuel cycle.

Firstly, the common features of the national approaches will be discussed.

The countries participating in the Joint Study include a transition phase in their development program before the full scale deployment of a nuclear energy system with fast reactors and a closed fuel cycle. The transition phase includes the deployment of an experimental fast reactor, then the instalment of a demonstration or prototype unit, before constructing a commercial size fast reactor.

Two countries are more advanced in this transition phase, i.e. France[6] and Russia, as they have already built and operated an experimental and a prototype fast reactor. India and Japan are operating successfully their experimental fast reactors since 1985 and 1977 respectively.

Most countries, other than Russia who operated a prototype (BN-350) successfully from 1973 till 1999, plan to start up a prototype reactor around 2020 to 2030, and commercialization of fast reactors is foreseen in a time period from 2020 to 2050.

As coolant for the prototype fast reactor all countries chose sodium in their reference design and, with the exception of Korea (that considers metal fuel in their reference design), they have selected MOX fuel for their prototype reactor. Russia and France consider also the possibility of other primary coolant options. Russia considers Pb and Pb-Bi and France gas as reactor coolant. Regarding the nuclear fuel almost all countries consider additional variants of the reference design, i.e. fuel with higher densities of fissile material such as nitrides, carbides or metal fuel that enable higher conversion (or breeding) capabilities.

[5] Canada and Ukraine participated in the discussions during the Joint Study but did not contribute to the assessments themselves.

[6] France operated a demo prototype and operated a commercial FR but intends to build new versions around 2020 to 2030.

Table 3.3. Comparison of national approaches to the development of a nuclear energy system with FR and closed fuel cycle

Countries	**Stages of FR development and reactor configuration**			**Coolant**		**Fuel**		**Fabrication / reprocessing**		**Some specific features of a long term approach**
	Experimental	**Demo prototype**	**Commercial size**	**Ref.**	**Var.**	**Ref.**	**Var.**	**Ref.**	**Var.**	
China	CEFR construction	2020-2025	2030-2035	Na		MOX	UOX Metal	Pellet/ advanced aqueous	Injection casting/	High breeding, on-site fuel cycle, MA-burners &ADS
France	Rapsodie 1967-1983	Phenix 1974 new FR 2020+	Super-Phenix 1986-1996, serial 2030+	Na	Gas	MOX	Car-bide	Pellet/ advanced aqueous	Plates (for Gas FR)	Break even cores, MA incineration, hydrogen production
India	FBTR 1985	PFBR construction	First serial 2020	Na		MOX	Metal	Pellet/ advanced aqueous	Sol-gel microsphere pellets/pyro	High breeding, co-location of FC facilities, thorium fuel
Japan	Joyo1977 operation, loop type	Monju 1994, JSFR demo 2025 loop type	First serial 2040-2050 loop type	Na		MOX		Pellet/ advanced aqueous		Low Breeding, Loop type reactor configuration
Rep. of Korea		2030 assumed for JS	Pool type	Na		Metal		/Pyro		Break even cores, reduction of SNF
USSR/ Russia	BR5/10 1959-2002 BOR60 1969	BN350 1973-1999 UOX, loop type	BN600 1980 UOX; BN800 construction, serial 2020-2025	Na	Pb-Bi, Pb	MOX	Nitride	Pellet/ advanced aqueous	Vibro-packing/pyro	Heavy metal coolants, breeding and break even cores, modular designs

Ref. = reference design, Var. = variant of design considered

A majority of countries opted for an advanced aqueous reprocessing process and several are investigating in addition a pyro-chemical process as an alternative option (Korea has chosen this process for their reference design). The advanced aqueous processes utilize:

- Co precipitation of uranium and plutonium to avoid separated plutonium, thus resulting in an enhanced level of proliferation resistance, and
- the possibility to separate from the spent fuel minor actinides (MA) and some specific fission products (e.g., high heat, long lived, mobile) to keep them in the fuel cycle for transmutation, thereby significantly reducing the requirements on the final depository of high level nuclear waste.

The advantages of pyro-chemical processes compared to aqueous processes include increased flexibility to treat advanced types of fuel such as nitride or metal fuel and the high radiation resistance of the salts used in this process. Partitioning and transmutation of MA and fission products is also possible with this process.

So to summarize national fast reactor development programs with a closed fuel cycle, they are very similar, i.e. there is only a limited number of significant differences to be discussed in the following.

The main differences among the national programs are: the time schedule for implementing the program and specific features of the fast core designs, i.e. the breeding rates and the corresponding fuel design. These differences are caused, firstly, by the starting positions of the countries, i.e. the history of nuclear power in these countries, and secondly by their planned rate of growth of their nuclear power capacities.

France and Japan have established already in the past a large nuclear power program based on thermal reactors and an open fuel cycle[8] and have accumulated a large amount of plutonium (mostly in spent fuel). Both countries expect, however, only insignificant to moderate growth of nuclear power capacity in the future. Thus, besides assuring a sustainable energy supply, their main goal of introducing fast reactors is to reduce the volume and toxicity of high level radioactive waste. A moderate breeding ratio, e.g., design of break even cores is sufficient to reach this goal within a well established long term schedule.

Russia has the most experience within the area of fast reactors of all countries participating in the Joint Study. It also has a large nuclear power program in operation for a long time based on thermal reactors and a large amount of stored spent fuel. But, contrary to Japan and France, it expects a very high increase of national nuclear capacity in the near future. Thus, it intends to introduce fast reactors on a shorter time scale than France or Japan and considers core designs that allow also conversion rates significantly greater 1.0, i.e. breeding.

The Republic of Korea has a large nuclear power program in operation based on thermal reactors, and, has been accumulating a large amount of spent fuel. Similar to Japan, it expects a moderate increase of national nuclear capacity in the future. Consequently, it plans for introduction of commercial fast reactor systems with a priority on break even cores that enable reduction of high level waste within a somewhat longer time schedule.

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[8] Open or once through fuel cycle id defined by direct disposal of spent fuel. Both Japan and France have also partly reprocessed fuel from thermal reactors and produced MOX fuel.

India has significant experience with the design and operation of fast reactors and the associated fuel cycle technologies. It has a rather limited nuclear power program in operation based on thermal reactors and therefore a limited amount of spent fuel that can provide plutonium for fast reactor fuel. But it expects a rapid and large increase of nuclear power capacity in the near future. Thus, to overcome a possible shortage of fissile material it aims to introduce fast reactors on a commercial basis as quick as possible and to develop core designs with conversion rates as high as possible (> 1.3 to 1.6), i.e. breeding. As mentioned in Chapter 2, India also considers introducing thorium into the national nuclear fuel cycle later in the future; ^{233}U produced from thorium fuel in fast reactors (core blankets) is planned to be used in fuel for advanced heavy water cooled reactors.

China is in a similar situation regarding the use of nuclear power as India. It has a rather small nuclear power program based on thermal reactors in operation but intends also to increase the nuclear capacity significantly in the near future. China started its fast reactor development program a little bit later than India and is therefore aiming to introduce commercial fast reactors somewhat later than India. Similar to India it also gives priority to core designs with high breeding rates to avoid a possible shortage of plutonium.

The differences and commonalities of national nuclear power programs discussed above are summarized in Table 3.4.

Table 3.4. Differences and commonalities of boundary conditions for national approaches to a nuclear energy system of fast reactors with a closed fuel cycle

	China	**France**	**India**	**Japan**	**Korea**	**Russia**	**Optimal conditions for developing FR**
Change in energy demand	Very high	Low	Very high	Low	High	High	Very high
Nuclear Share	Small, fast growing	Very high, stabilizing	Small, fast growing	High, stabili-zing	High, growing	High, growing	Very high, fast growing
FR Technology Maturity	Experi-mental program	Commer-cialization	Demo	Demo	Design works	Commer-cialization	Commer-cialization
CNFC Techno-logy Maturity	Up-coming	Highly developed, FR MOX fuel mastered	Developed, FR carbide and MOX fuel demon-strated	Highly deve-loped, FRMOX fuel mastered	Up-coming	Highly developed, FR UOX fuel mastered, FR MOX demon-strated	Highly developed, UOX and MOX fuel mastered
Pu in SNF	Small	High	Small	High	Moderate	High	High

The first row, called “Change in energy demand” means the demand for power capacity in the future; “Nuclear share” means the share of nuclear power in the energy mix of the country; “FR technology maturity” and “CNFC technology maturity” describe the status of the development program of FR and fuel cycle technologies, respectively; “Pu in SNF” is the availability of plutonium in spent nuclear fuel. The last column of Table 3.4, presents the optimal conditions for establishing a fast reactor system with a closed fuel cycle as agreed by the participants of the Joint Study.

Looking at the country data in Table 3.4 a remarkable situation can be noted: no individual country participating in the Joint Study reflects all that factors that favour an optimal development of a nuclear energy system consisting of CNFC-FR but all countries when considered jointly do so. Low increment in energy demand in some countries will probably hamper full utilization of the potential of their developed CNFC-FR infrastructure while in countries with rapidly increasing energy demand the lack of the resources needed for a CNFC-FR system might constrain growth of nuclear power deployment.

Thus, data in the Table 3.4 indicate generic preconditions or opportunities for reaching mutual benefits by establishing multilateral approaches to a nuclear fuel cycle of nuclear energy systems with fast reactors.

3.3. Comparison of global and national scenarios of nuclear power development

After a detailed survey of a set of global scenarios of nuclear power development projected by different energy agencies and organizations, scenario B2 of Special Report on Emission Scenarios (SRES) of the Intergovernmental Panel on Climate Change (IPPC) [10] was selected by the participants of the Joint Study as consistent with their expectations of nuclear growth during the 21st century. The scenario approach helped to evaluate the role of the CNFC-FR in the most important area of the assessment, i.e. assurance of nuclear fuel supply.

An analysis of the natural uranium consumption by thermal reactors with the once-through fuel cycle (OTFC) was performed for the global SRES B2 scenario with the use of computer code DESAE [12]. This code was available for all participants of the study. The analysis showed that in case of growth of global nuclear capacities to 2000 GW_e by 2050 and 5000 GW_e by 2100 the annual uranium extraction would reach the level of 300 to 350 thousand tons per year by 2050, and 750 to 900 thousand tons per year by 2100. The cumulative uranium consumption by 2050 would exceed 10 million tons by 2050 and 40 million tons by 2100. This predicted consumption by 2100 goes beyond the sum of conventional[9] uranium resources of about 16 million tons in the price category of less than 130 US$/kg U estimated in Ref. [11]. Unconventional uranium resources are estimated to be around 22 million tons for uranium mainly in phosphates with estimated production costs around 130 US$/kg U, and in seawater about 4 billion tons however with production cost in the range of several hundred US$/kg U.

Thus, looking at global scenarios of demand and supply of natural uranium the countries involved in the Joint Study concluded that a global nuclear energy system based only on thermal reactors and open fuel cycles could run out of uranium fuel in the price category of less than 130 US$/kg U sometime towards the end of the 21st century. To avoid increases of fuel prices it will be necessary to introduce commercial fast reactors with a closed fuel cycle in time to assure the sustainability of such a global nuclear energy system. The fuel for the start up of fast reactor systems will have to be based primarily on plutonium recycled from spent fuel of thermal reactors.

[9] Conventional resources are defined in Ref. [11] as resources from which uranium is recoverable as a primary product, a co-product or an important by-product. Unconventional resources are resources from which uranium is only recoverable as a minor by-product, such as uranium associated with phosphate rocks, non-ferrous ores, black schists, and lignite.

Spent fuel from thermal reactors typically contains 6 to 12 g Pu per kg of spent fuel, i.e. about 1 %. The total amount of spent fuel and separated Pu is expected to reach globally about 450000 t_{HM} and 120 t_{Pu} respectively by 2020. Thus, on a global basis the scenarios indicate that enough spent fuel will be available that can be used to produce firstly the necessary fast reactor fuel in time and thereafter fuel for thermal reactors as well based on plutonium produced in the fast reactor systems. Such nuclear energy systems could be run several hundred years without running out of fuel (further discussed in Section 4.5).

However, national scenarios (see Table 3.4 above) also indicate that, in particular, countries with very large increments of nuclear power demand might not have enough plutonium domestically available on a national basis to start up the full size of their planned fast reactor system. But, by taking the global situation of nuclear power into account as discussed above, this possible national shortage of plutonium could be mitigated via a multi national approach [13].

CHAPTER 4
ASSESSMENT OF A CNFC-FR SYSTEM USING THE INPRO METHODOLOGY

4.1. Definition of a reference model of a fast reactor system

Within the Joint Study a reference model of a nuclear energy system consisting of fast reactors (FR) with a closed fuel cycle (CNFC) was developed. The main characteristics of this reference model are presented in Table 4.1 (reactor design) and Table 4.2 (fuel design). The model is based mostly on characteristics of operating fast reactors and data of evolutionary designs in the countries participating in the Joint Study.

Table 4.1. Specification of the reference model of a FR

Parameter	Reference data	National variant
Power	1000 MW_e	1500 MW_e (Japan).
Coolant	Sodium	
Reactor configuration	Pool	Loop (Japan).
Power plant	With minimum 2 reactor units.	
Thermal efficiency	43 %	> 39 % (Korea).
Capacity factor	85 % - 90 %	
Life	60 yrs	
Fuel	MOX	MOX in first development stage and metal in next stage (China, India, and Korea).
Construction time	54 months (from concrete pour to first criticality).	
Breeding ratio	1.2	1.0 (Korea, France, and Japan).
Minor actinides recycling	None	Yes (France); optional (Japan, Russia, Korea, China, and India).
Seismicity	Parametric (depending on region).	
Burnup	150 GWd/t (average).	120 GWd/t (metal, Korea).
Specific steel consumption	3.5 t/MW_e	

In the following background is provided for the chosen parameters in Table 4.1 above.

To exploit the economy of scale a *reactor capacity* of 1000 MW_e is chosen and at least two units are assumed to be built on the same site. Based on existing large experience with sodium as *coolant* of FR in a pool type *configuration*, the chosen reference design of the reactor is a pool type arrangement and includes a primary and secondary *coolant* loop with sodium as the coolant.

As an option the reference plant with at least two[10] *FR units* of 1000 MWe utilizing a closed fuel cycle is assumed to be *co-located to fuel fabrication*, *reprocessing and waste management facilities* of matching capacities in a single physical protection boundary. With such a configuration time delays and proliferation risks, associated with transportation of fuel, could be significantly reduced. After the first few years of operation, the CNFC-FR system is assumed to be more than self-sustaining regarding inventories of Pu and the only input required will be make-up uranium to compensate its consumption by fission and breeding processes.

Increasing *thermal efficiency* decreases unit energy cost and reduces the thermal pollution (waste heat) levels of the environment. In the past for a demonstration type FR, a steam temperature of around 766 K was selected in order to restrict the hot pool sodium temperature to 820 K at nominal power. Based on accumulated experiences of material data and revised high temperature design code rules (standards), it seems possible, in the future, to operate at higher hot pool sodium temperatures generating steam at 803 – 813 K, thus, matching the steam conditions of fossil fuelled power plants. These higher primary temperatures will lead to increased plant efficiency together with less thermal pollution. Two fast reactors are currently operating with a thermal efficiency of 43 to 45 % that is the highest value of all operating nuclear power plants [15]. Hence, a minimum plant efficiency of 43 % should be the target.

Over 300 years of operational experience of fast reactors worldwide provides confidence that high *capacity* factors can be obtained in future CNFC-FR systems. Rout causes of technical problems that occurred in the first generation FR (e.g., MONJU and PHENIX) have been identified. Based on this experience fast reactor technology has reached sufficient maturity eliminating potential causes of defects and offering technological solutions, for example how avoiding steam generator leaks causing sodium fires. Hence, capacity factors around 85 % to 90 % should be the target.

Based on detailed studies on material data and structural mechanics considerations, a 60 year *design life* seems possible to be achieved. But, long time material degradation mechanisms such as corrosion, erosion, and irradiation effects are to be crucially evaluated, tested and assessed.

Based on the existing capabilities of nuclear industry and economic considerations, mixed *oxide (MOX) fuel* is the first choice for FR. The major advantage of oxide fuel is the capability to achieve high burnups as its degradation mechanisms are well understood and adequate designs have been developed. A disadvantage of oxide fuel is the limitation of achievable breeding rates (< 1.2). Thus, in considering a high growth rate of energy demand, especially in some Asian countries, to achieve high breeding rates metal fuel is identified as an option.

Construction time of FR could be definitely shortened based on international experience. It has been demonstrated that reactors of 500 MW_e – 1000 MW_e can be constructed in about five years. Based on the development of modern construction technology and efficient project management techniques, 54 months is considered to be an achievable parameter.

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[10] The Joint Study concluded that NFC facilities collocated to 4 fast reactors could be an economically viable option.

The *breeding ratio* targeted for oxide fuel is 1.2 and for metal fuel should be around 1.6. Suitable core designs of fast reactors for burning of actinides and other long lived and/or high heat producing fission products should be developed.

Enhancing the target *burn-up* of the FR fuel has a direct effect on reduction of unit energy cost. Based on large experience, especially in the U.S., Russian Federation and France, a burn up of 150 GWd/t should be considered in the initial phase with an ultimate target of 200 GWd/t. To achieve this ultimate target the selection of advanced materials and improvement of present materials through characterization, design of materials, and modelling of material behaviour is necessary.

Table 4.2. Specification of the reference model of fuel cycle facilities of CNFC

Parameter	**Reference model data**	**National variant**
Fuel Fabrication	Pellet by powder metallurgy route.	Vibro compaction (Russia), Injection casting (Korea, China, and India).
Reprocessing	Advanced aqueous process with Pu loss < 0.05 % and minor actinide partitioning. U & Pu oxides to be co-precipitated in co-located and optimized reprocessing, re-fabrication and waste management facility for a number of reactors (variable) at the same site.	Pyro-processing (Russia, Korea, India, and China).
Cooling time before reprocessing	4 years.	1 year (China, India, and Korea).
Solvent used in reprocessing	Tri-n-Butyl Phosphate or homologue.	
U-Pu separation in reprocessing	Co-processing with no Pu separation.	
Plant life of reprocessing	40 years.	
High level waste management	Glass vitrification.	
High Level Waste (Volume/MW_e)	To be developed.	

For the reference CNFC-FR system capacities of fuel fabrication and reprocessing plants are much lower compared to thermal neutron reactor systems because of the very high burn-up achieved in a FR. To become an economically viable option the reference fuel cycle facilities should have a capacity to meet requirements of four reactors of 1000 MW_e each, and the quantity of discharged fuel from each reactor is assumed to be five tons heavy metal (HM) per year. Thus, for a park consisting of four FR, the fuel cycle facility should have the processing capacity of 20 tons of HM per year.

Optimizations of the plant sizing would be needed primarily for reprocessing plants as it requires heavily shielded space due to higher burn-up (compared to thermal reactor systems) of fuel being processed and subsequent higher activity handled.

In this context, it may be mentioned that the concurrent solvent extraction technology (using centrifugal extractors) does not need the huge headspace or large footprint required by the earlier generation equipments (pulse-columns and mixer-settlers). However, significant design restraints arise from the criticality point of view, as a high surface to volume ratio is required for plutonium bearing solutions in slab tanks or annular tanks. Use of borated (neutron poisoned) steel as structural material for tanks as well as tanks with poison tubes should result in higher holding capacities.

The *fuel fabrication* for the CNFC-FR system should be based on the mixed powder route. Mixed oxide could be made by co-processing and co-precipitation and this mixed oxide product may be suitably diluted by adding UO_2 powder to make the fuel for multiple compositions of FR core. Since U–Pu separation is not envisaged, several process steps are eliminated resulting in a reduced number of process equipment, tankage and operations leading to significant reduction in the processing cost.

The advanced *reprocessing* operation of the reference plant involves recovery of unused and bred fissile materials as well as recovery of minor actinides (MAs) and selected high heat producing or long-lived fission products (LLFP) in a form suitable for immediate recycling in the reactor or co-located transmutation systems. It is assumed that advanced aqueous processes can be used for the tentative burn-up of 200 GWd/t and a 360 days *cooling* period of the discharged fuel.

The *solvent extraction process* involved will be an advanced PUREX process. Advanced PUREX process is a subset of the classical PUREX process with several novel features envisaged. Its radical approaches aim at combinations of waste treatment processes to achieve a significant cost saving. The main features of the advanced PUREX process are:

- A high efficiency extractor (centrifugal extractor) will be used in the solvent extraction operation enabling a small residence time to limit radiation-induced solvent/diluent degradation and zirconium extraction. Smaller holdup and lower inventory are the added benefits. Utilization of centrifugal extractors (CE) will also enable highly economical civil construction for the reprocessing facility as CE do not require tall headspace leading to a significantly reduced height as well as volume of the shielded radioactive cell;

- *U-Pu separation* is not envisaged resulting in economic benefits and increased proliferation resistance; and

- The primary tri-n-butyl phosphate (TBP) or its homologue based extraction could be used for the additional separation of Np, tritium and ^{99}Tc in the high activity (HA) cycle itself. Tritium and ^{99}Tc may be reclaimed from the aqueous nitric solutions by additional chemical processes. By confining the tritium to the HA cycle, a plant wide contamination by tritium is avoided. ^{99}Tc could be converted to metal form and irradiated at peripheral positions of FR cores for transmutation. Similarly, another long lived fission product ^{129}I could be absorbed in caustic solutions and the resulting compound NaI could be irradiated in transmutation systems.

It may be mentioned that in the aqueous route of reprocessing, extremely high separation factors (also called decontamination factors) of 107 and high recovery rates ≥ 99.8 % are routinely achieved. For the reference CNFC-FR system the stipulated Pu recoveries are 99.95 % or more. Emphasis is on robust online analytical techniques for monitoring of Pu in raffinate streams and activity levels in the loaded organic streams and subsequent fine tuning

of operating parameters for a significant reduction in the rework of off-grade products. This should also result in increased plant throughput and availability.

To reduce the long term radiotoxic inventory of the final depository separation and transmutation of MA and long lived fission products (e.g., ^{99}Tc) is developed; to minimize the heat load in the final depository fission products such as ^{137}Cs and ^{90}Sr with high heat production could also be separated from spent fuel and transmuted in the reactor.

Recently, several new extractants have been reported. To achieve actinide-free status for high level waste, recovery levels of MA are assumed to be 99.9 %.

For the reference CNFC-FR system, the solid waste volumes shall be equal or less the accepted benchmarks. In addition additional waste reduction features like acid recovery/recycle and online solvent purification by improved methods like vacuum distillation would significantly reduce the waste volumes produced.

The *reprocessing plant* should have a *design life* of 40 years, comparable to the reactor lifetime. Lifetime extension seems possible by periodic condition monitoring of the plant components performing in-service inspection by remotely controlled devices like robot mounted cameras, special fibre-optic probes for visual inspection and ultrasonic measurements for wall thinning.

In the *HLW management* plant, the main problem of corrosion is due to molten glass. In this context, relevant parts of plant are to be designed for remote maintenance and replacement. Advanced technologies like cold-crucible technique should reduce the effect of corrosion and ensure long life with economic benefits.

Construction time of 72 months for the fuel cycle facilities is assumed from the first pouring of concrete to active commissioning. Modern project scheduling, management and erection methods may reduce it to 60 months minimum. It is essential to complete construction and hot commissioning of the fuel cycle facility along the reference reactor so that discharged fuel could be processed immediately after the short cooling period and out of pile inventory of fissile material could be kept to a minimum technically achievable.

This reference model of a CNFC-FR system and, in some cases the national variant of it, were assessed using the INPRO methodology as discussed in the following sections.

4.2. Assessment of a CNFC-FR system in the INPRO area of economics

INPRO has developed one basic principle in the area of economics asking for nuclear energy and related products to be competitive in comparison with alternative energy sources available in a country. The corresponding user requirements demand, firstly, that to be sustainable in a country or region products of nuclear energy (electricity or heat) should be cost competitive with the cost of locally available alternative energy sources (user requirement UR1) such as renewables (hydro, solar, wind, etc.) or fossil plants, and, secondly, that the total investment funds required to design, construct and commission a nuclear energy system can be raised and the risk of investment is acceptable compared to investments into alternative energy projects (UR2 and UR3). UR4 is directed at technology developers and it asks that new nuclear energy systems have flexibility to meet requirements of different markets.

The Joint Study concluded that currently operating sodium-cooled fast reactors and closed fuel cycles cannot compete economically against alternative energy sources including thermal reactors, primarily because of the high upfront investment costs of both the reactor and the

fuel cycle facilities. Capital costs of the “first of a kind” fast reactors were 40 % up to three times higher than capital cost of thermal reactors. However, the fast reactor and related fuel cycle systems under development in the participating countries promise decisive cost reductions (see for example Figure 5.2 in Chapter 5) within the next 10 to 20 years by improvements in the design, such as design simplification, reduction of steel consumption by reducing number of loops and wall thickness of main components, elimination or reduction of size of reactor systems, more compact plant layout, increasing fuel burnup, increasing thermal efficiency and load factor, serial construction with reduction of time of construction, etc.

Within the Joint Study five of the participating countries, i.e. France, India, Japan, Republic of Korea, and the Russian Federation performed a detailed economic INPRO assessment of sodium cooled fast reactors with a closed fuel cycle to be deployed within their national borders. The five assessors evaluated their national design of a fast reactor system (but not the reference model) under development and took country and design specific boundary conditions for an INPRO economic assessment into account, such as size of output of the fast reactor, construction time, discount rates, and cheapest alternative energy source available in the country, as shown in the following Table 4.3.

Table 4.3. CNFC-FR system parameters of national approaches and of the reference model relevant for economics

CNFC-FR parameters relevant for economics	**France**	**India**	**Japan**	**Korea**	**Russia**	**Reference model**
Fast reactor power ($MW_{e)}$	1500	500	1500	600	1800	1000-1500
Time of construction (yr)	4	5	7	2.7	6	4.5
Discount rate (%)	8	12	2	8	5	-
Cheapest alternative energy source.	GCC	PHWR	ALWR	GEN IV	FFPP	-

GCC=gas combined cycle power plant; PHWR=pressurized heavy water reactor; ALWR=advanced light water reactor; FFPP=fossil fuelled power plant; GEN IV= Generation IV metrics

Based on results of a French study between 5 to 15 % of the fast reactor capital costs can be compensated by the cheaper fuel costs of a fast reactor system in comparison to a thermal reactor system. All national nuclear energy systems with fast reactors and a closed fuel cycle once developed were found to be competitive regarding cost of electricity against the cheapest available national alternative energy source (see Table 4.4), i.e. they will be capable of producing electricity cheaper than the national alternatives, thereby fulfilling the first INPRO economic user requirement UR1.

Table 4.4. Results of INPRO assessment of costs of electricity of national CNFC-FR

Country	Electricity cost of CNFC-FR (mills $/kWh)	Electricity cost of cheapest alternative energy source (mills $/kWh)
France	35.41	44.07
India	41.00	45.00
Japan	15.10	26.59
Korea	31.15	34.00
Russia	17.74	24.50

The ability to finance a CNFC-FR system based on the financial figures of merit (e.g., internal rate of return, IRR, return of investment, ROI) and availability of national funds was confirmed for all five national systems, thus fulfilling the second INPRO economic user requirement UR2. The following Table 4.5 shows the superiority of financial figures of merit of fast reactors in comparison to the cheapest available alternative energy source and confirms the availability of national funds.

Table 4.5. Comparison of financial figures of merit of national CNFC-FR with cheapest alternative and of funds needed for CNFC-FR with national available funds for such a project

	France	India	Japan	Korea	Russia
IRR of CNFC-FR (%/year)	14.0	16.2	19.0	43.6	6.5
IRR of alternative (%/year)	10.0	10.0	10.0	8.0	4.0
ROI of CNFC-FR (%/year)	14.0	25.5	12.0	169.1	-
ROI of alternative (%/year)	10.0	10.0	3.0	8.0	-
Funds needed for CNFC-FR ($M)	-	1330	1615	807	2410
Funds available in country ($M)	-	1490	2596	957	3500

The risk of investment into a CNFC-FR system characterized by status of licensing, construction schedules, and political environment (long term commitment of government), and by robustness of economical figures against possible changes in boundary conditions was found acceptable by all assessors, thereby fulfilling INPRO economic user requirement UR3.

The overall conclusion of the INPRO economic assessment is that a nuclear energy system consisting of a series of fast reactors incorporating improvements to be developed within the next 10 to 20 years will meet INPRO's economic basic principle, i.e. the nuclear energy system CNFC-FR *will be affordable and available* in 10 to 20 years in the countries mastering this technology.

4.3. Assessment of CNFC-FR in the INPRO area of infrastructure

INPRO has defined one basic principle in this area calling for a limitation of the effort necessary for establishing (and maintaining) an adequate infrastructure in a country that intends to install (or maintain or enlarge) a nuclear energy system. This should be achieved by regional and international arrangements that should be made available to such countries. The corresponding INPRO user requirements (UR) recognize the need for establishing and maintaining a national legal framework including international obligations, the need to define the necessary industrial and economic infrastructure for a nuclear power program, the need to lay out the appropriate measures to secure public acceptance, and the need to address the availability of adequate human resources. Contrary to all other INPRO areas that, in general, address the designer/developer of an nuclear energy system, in the area of infrastructure the

INPRO requirements are directed primarily to the government, the operator of a nuclear facility, and to national industry.

All participating countries of the Joint Study have a well established nuclear power program based on thermal reactors and some operate (or will operate soon) prototypes of fast reactors[11]. To successfully operate their nuclear power programs each country has an adequate infrastructure in place for a long time.

Six of these countries, i.e. China, France, India, Japan, Republic of Korea, and the Russian Federation performed an INPRO assessment of their national infrastructure using the INPRO methodology to check its adequacy also for operating a nuclear energy system with FR and CNFC. Each INPRO criteria was evaluated and full agreement with the acceptance limits was confirmed as outlined in the following.

For the existing national nuclear power systems an adequate legal framework is established in accordance with international standards and institutional arrangements (e.g., regulatory bodies) for radiation protection and safety are in place, thus, INPRO user requirement UR1 (legal and institutional infrastructure) is fulfilled.

The financing of the introduction of CNFC-FR systems is available, the capacity of the FR match the national energy demand and supply forecast, and the national industries of the six countries are capable of manufacturing, constructing and operating of such nuclear energy systems. The additional investment in infrastructure needed for a CNFC-FR system is compensated by the benefits gained by the establishment of such a system, thus, user requirement UR2 (economic and industrial infrastructure) is met.

Each country has a national energy plan established that includes the role of nuclear power and has communicated it to the public appropriately to assure public acceptance. Operational data of nuclear power plants are available on the internet. In all countries there is a long-term commitment of the government to nuclear power, thus, user requirement UR3 (socio-political infrastructure) is fulfilled.

Sufficient human resources needed for installing and operating a CNFC-FR system are available via the existing educational and training institutions, thus, user requirement UR4 (infrastructure of human resources) is fulfilled.

Nevertheless, infrastructure is a critical issue for a CNFC-FR system, especially, if deployed in a country that is not a developer of this technology. Even within the participants of the Joint Study a survey has shown complementarity of national conditions that creates prerequisites for mutually beneficial long term collaboration. No individual country, taken separately, displays the full set of factors favouring development and deployment of a CNFC-FR system such as high energy demand, high level of nuclear technology and infrastructure maturity, high resources of spent nuclear fuel, etc. (see also Table 3.4 in Section 3.3).

The Joint Study came to the conclusion that the assessed CNFC-FR technology is well suited for and might require a regional or multilateral approach. Such an approach will assure the availability of the corresponding front and backend of fuel cycle services to technology holder as well as to technology user countries within a global or regional nuclear architecture.

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[11] Details of fast reactor programs were presented in Chapter 2.

4.4. Assessment of CNFC-FR in the INPRO area of proliferation resistance

Proliferation resistance (PR) of an NES consists of a combination of intrinsic features, i.e. technical design characteristics such as easiness of inspection, and extrinsic measures, i.e. commitments of States such as safeguard agreements. INPRO has produced one basic principle (BP) in this area of PR that requires intrinsic features to be implemented always together with extrinsic measures in an NES throughout the full life cycle. The corresponding INPRO user requirements ask the State to establish and maintain a sufficient legal framework, and the designer to keep the attractiveness of nuclear material (NM) low, make diversion of NM difficult and easy detectable, incorporate multiple barriers, and implement cost effective safeguard measures.

Instead of directly applying the INPRO methodology on a CNFC-FR system the Joint Study investigated the differences in regard to proliferation resistance between a nuclear energy system consisting of thermal reactors with an open or once through fuel cycle (OTFC) and a fleet of fast reactors (FR) with a closed nuclear fuel cycle (CNFC). It was concluded that a CNFC-FR system will have several features that increase the proliferation resistance in comparison to an OTFC system with thermal reactors.

First of all a CNFC-FR system will not need enrichment as the fissile material for fresh fuel, i.e. plutonium is produced via transformation (neutron capture) of ^{238}U and made available via reprocessing[12].

Further, to avoid the significant proliferation risk of currently operating reprocessing facilities that produce separated plutonium, the envisaged advanced reprocessing technologies to be used in a CNFC-FR system will always keep uranium and plutonium in a compound. Two types of such reprocessing technologies are being developed: the advanced aqueous and the pyrochemical reprocessing. Pyroprocessing facilities are compact and could together with a fuel fabrication facility be collocated with reactors thereby eliminating the need to transport fuel resulting in an increase of proliferation resistance. Advanced aqueous reprocessing facilities are more suited for centrally operated large scale plants. If such a plant would be an international institution (called a multinational fuel cycle centre) it would meet proliferation concerns and at the same time guarantee nuclear fuel supply to states having no access to the reprocessing technology.

Additionally, as discussed in more detail in Section 4.6, certain minor actinides and fission products in spent fuel could be recycled and added to fresh fuel leading to high radiation levels. A high radiation level of fresh fuel is regarded as an intrinsic feature increasing proliferation resistance.

Reprocessing also reduces the proliferation risk of large quantities of stored and disposed spent nuclear fuel containing Pu (“Pu mines”) that are currently accumulated in thermal reactor systems with open fuel cycles.

Within the Joint Study the Japanese program for developing a nuclear energy system of FR with CNFC (called FaCT) was assessed in detail using the INPRO methodology for this area. All INPRO criteria were found by the assessor to be fulfilled.

[12] Similar arguments could be made for a thorium/uranium cycle.

It has to be mentioned that there are also some features of CNFC-FR systems that lead to a decrease of proliferation resistance compared to thermal reactor systems with an OTFC. One example is the much higher content of fissile material in FR fuel compared to thermal reactors, which makes this nuclear material more attractive for diversion. Another factor is the increased transport volumes for fuel that will be required; this could also lead to problems in the area of security and public acceptance.

4.5. Assessment of CNFC-FR in the INPRO area of environment

In the INPRO area of environment two aspects are covered, namely outputs from a nuclear facility, which represent environmental stressors; e.g., discharges of radionuclides, and inputs to a NES which may lead to depletion of natural resources, such as uranium, zirconium, etc.

Consequently, INPRO has developed two basic principles (BP) in this area. BP1 calls for acceptability of environmental impacts caused by nuclear facilities on humans and the environment and BP2 requires the confirmation of the long term availability and optimized use of material resources needed to operate a nuclear system. The two INPRO user requirements (UR1.1 and UR1.2) corresponding to environmental BP1 ask for environmental stressors, e.g., release and impact of radioactive substances from a nuclear facility, to be within the relevant[13] standards (e.g., national regulatory limits) and the application of the ALARP concept[14].

The first INPRO UR2.1 related to second basic principle BP2 asks for availability of fissile and fertile materials needed for fabrication of nuclear fuel and of materials needed for construction and operation of nuclear facilities for a period of at least hundred years, and an improved usage of such materials compared to operating nuclear systems in 2004. The second UR2.2 (related to BP2) asks primarily for an adequate energy output in comparison to the energy needed to construct and operate the nuclear system.

To be able to assess nuclear energy systems in regard to their impact on the environment, i.e. via releases of toxic elements and depletion of resources, as a first step an analysis was performed within the Joint Study. Six nuclear energy systems (NES) were analyzed, all with a constant power capacity of 60 GW_e (and about 400 TWh_e output per year) during the 21st century [14]; the first five NES are “steady state” scenarios and No.6 is a dynamic scenario. The six NES analyzed were:

- NES No.1: A PWR fleet with UOX fuel utilizing an open fuel cycle (Spent nuclear fuel sent to repository);
- NES No.2: A PWR fleet, with spent UOX fuel reprocessing, vitrification of fission products (FP) and minor actinides (MA), and Pu "mono"-recycling (spent MOX fuel sent to interim storage);
- NES No.3: A PWR fleet, with reprocessing of all spent UOX and MOX fuel, Pu recycling in MOX assemblies, and vitrification of FP and MA. At equilibrium, the fleet is composed of 74 % of PWR loaded with UOX, and 26 % loaded with MOX;

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[13] The term “relevant” means at the time of installation of a new nuclear facility.

[14] ALARP means “as low as reasonable practicable”. This concept is described in more detail in Section 4.3.2 of Volume 1 of Ref. [2].

- NES No.4: A mixed fleet with 45 % of PWR, 55 % of fast reactors (FR) recycling Pu and incinerating 90 % of americium in transmutation targets. Neptunium and curium are vitrified with FP;

- NES No.5: A FR fleet recycling all MA together with plutonium (fully closed cycle). Only FP are vitrified; and

- NES No.6: A reactor fleet starting from a pure PWR fleet at the beginning of the 21st century, after 2020 PWR are gradually replaced by EPR and after 2035 by FR (see Figure 4.1), and it becomes a pure FR fleet with a closed fuel cycle at the end of the 21st century.

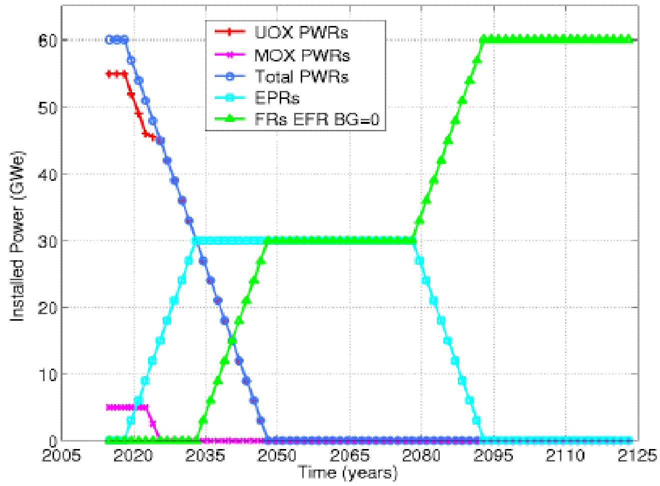

Figure 4.1. Dynamic nuclear energy system NES No. 6 analyzed in the Joint Study [16].

For these six NES two types of analysis[15] were performed determining several environmentally significant parameters:

- A life cycle analysis (LCA) using the TEAM software determining the consumption (per TWh_e produced) of natural uranium, but also of oil, gas and coal, the equivalent tons of CO_2 produced (greenhouse effect potential impact), and emissions of CO_2, CH_4, N_2O, NO_x, SO_x, and particles (Table 4.6 and 4.7); and

- An evaluation of public exposure (mSv per year) for all facilities of the nuclear fuel cycle, i.e. mining/milling, conversion, enrichment, UOX and MOX production, power plants, reprocessing, low level waste storage, interim storage, and high level waste disposal. The dose values presented are based on experience with existing fuel cycle facilities in France and on extrapolation thereof for innovations such as advanced reprocessing or fast reactors (Table 4.8).

[15] A third analysis was performed for the six NES that determined parameters relevant for the area of waste management and is presented therefore in the following Section 4.6 (waste management).

Table 4.6. Results of life cycle analysis with TEAM software in regard to resource depletion

	Unit/ TWh_e	NES No.1	NES No.2	NES No.3	NES No.4	NES No.5	NES No.6		
							2015 - 2035	2035 - 2095	> 2095
Oil consumption	Ton	600	600	600	400	100	600	350	100
Coal consumption	Ton	1 000	1000	900	700	400	1000	650	400
Natural gas consumption	Ton	400	400	400	200	100	400	250	100
Natural uranium consumption	Ton	23.4	20.9	20	10.6	1.7	22.2	12.5	1.7

Table 4.6 above clearly demonstrates the significant reduction of resource depletion (especially natural uranium) in a fast reactor system (NES No.5 and NES No.6 after 2095) with a closed fuel cycle in comparison to a thermal system (NES No.1) with an open fuel cycle. Mono or even multi recycling of Pu in thermal reactors (as in NES No.2 and No.3) does not significantly reduce resource depletion compared to NES No.1. A mixed fleet of thermal and fast reactors (NES No.4) produces results that are of course between NES No.3 and No.5.

Based on the analysis results it was stated that in regard to basic principle BP2 (resource sustainability) NES No.5 (and NES No.6 after 2095) with FR and a closed fuel cycle is the optimal solution, as it drastically reduces the amount of natural uranium consumption and enables practically unlimited operation of the NES. Therefore, the Joint Study suggests, such an energy system could be even called a *de facto* renewable one.

Table 4.7. Results of life cycle analysis with TEAM software in regard to environmental impact of non radio active releases

	Unit/ TWh_e	NES No.1	NES No.2	NES No.3	NES No.4	NES No.5	NES No.6		
							2015 - 2035	2035 - 2095	> 2095
Greenhouse effect potential impact	CO_2 eq. Ton	4600	4600	4400	3000	1400	4600	2900	1400
CO_2 emission	Ton	4400	4400	4200	2900	1400	4400	2800	1400
CH_4 emission	Ton	6	6	6	4	2	6	3,5	2
N_2O emission	Ton	0.1	0.1	0.1	0.1	0.02	0.1	0.06	0.02
NO_x emission	Ton	15	15	14	10	4	15	10	4
SO_x emission	Ton	24	25	24	16	7	24	15	7
Particles emission	Ton	41	36	35	19	2	38	22	2

Table 4.7 above demonstrates the reduction of releases (gaseous and solid) to the environment of non radioactive elements in a fast reactor system compared to a thermal system. The impact of the gaseous releases are summarized by the so called greenhouse effect expressed in a CO_2 equivalent ton per TWh_e.

Table 4.8. Results of the evaluation of environmental impact of release of radioactive nuclides

	Unit	NES No.1	NES No.2	NES No.3	NES No.4	NES No.5	NES No.6		
							2015 - 2035	2035 - 2095	> 2095
Mines and processing	mSv/yr	< 1 (in the less favourable case of realistic scenario)							
Conversion	mSv/yr	Pierrelatte level ~0.002 Malvési level ~0.07				-	Pierrelatte 0.002 Malvési ~0.07		-
Enrichment	mSv/yr	Eurodif level ~0.0006					Eurodif level ~0.0006		
UOx fabrication	mSv/yr	Romans level ~0.0003					Romans level ~0.0003		
MOX fabrication	mSv/yr	-	Melox level ~10^{-5}						

Table 4.8. Results of the evaluation of environmental impact of release of radioactive nuclides (continued)

	Unit	NES No.1	NES No.2	NES No.3	NES No.4	NES No.5	NES No.6
Nuclear power plant	mSv/yr	~0.001					
Reprocessing and advanced reprocessing	mSv/yr	-	La Hague level = 0,01 Advanced reprocessing : < 1 for future facilities (current regulation) + COGEMA commitment < 0,03				
Low level waste storage	mSv/yr	2 10^{-3} in operating phase ; 8 10^{-3} in surveillance phase					
Very low level waste storage	mSv/yr	9 10^{-3} operating phase ; 9 10^{-5} in surveillance phase					
HLW deep geological disposal	mSv/yr	Long term impact < 0.25 (Safety Fundamental Rule)					
Interim storage	mSv/yr	Glass canister storage at La Hague < 10^{-3}					

Table 4.8 above demonstrates that all six NES investigated produce a public radiation dose far below the current regulatory limit of 1 mSv/yr for public exposure. Thus, all NES studied have no significant radiological impact on the environment and must be classified as comparable in this aspect.

Based on the analyses above the assessor concluded that all NES evaluated are clearly fulfilling the INPRO environmental basic principle BP1 (acceptability of expected adverse environmental effects).

Evaluation of global U demand and supply performed by Japan

In addition to the NES evaluated above that is representative for a single industrialized country like France, in the Joint Study an evaluation of the global demand and supply of uranium was performed by Japan. As a reference scenario SRES B2 was selected predicting a demand of 5000 GW_e of installed nuclear capacity at the end of the 21^{st} century. The total

conventional resources[16] of uranium (in the price category of less than 130 US$/kgU) available after 2005 were defined as 15 million tons based on Ref. [11].

Installing only thermal (LWR and HWR) reactors with a closed fuel cycle (Pu recycling) would lead to consumption of all defined conventional uranium resources (15 million tons) by about 2070 and to a maximum capacity of about 1000 GW_e around 2040 decreasing to only 290 GW_e by 2100.

Introduction of fast reactors with high breeding ratios (1.2) around 2020 to 2030 would enable to reach the defined 5000 GW_e between 2110 and 2125, i.e. leading to a shortage of 360 GW_e to 1600 GW_e installed nuclear capacity in 2100. The least power shortage is achieved if the ex-core time of spent fuel (cooling, reprocessing and fabrication) is reduced from 5 to 3 years.

4.6. Assessment of CNFC-FR in the INPRO area of waste management

The four INPRO basic principles have been derived from the nine IAEA Fundamental Principles of Radioactive Waste Management. Thus, the generation of waste shall be kept by design to the minimum practicable, waste shall be managed so as to secure an acceptable level of protection of human health and the environment regardless of the time or place at which impacts may occur, waste shall be managed in such a way that undue burdens are not imposed on future generations, and interdependencies among all waste generation and management steps shall be taken into account. These principles in turn lead to INPRO requirements to minimize the generation of waste with emphasis on waste containing long-lived toxic components that would be mobile in a repository environment, to limit exposures of humans to radiation and chemicals from waste, to limit releases to the environment of radio nuclides and chemical toxins, to specify a permanently safe end state for all wastes and to move wastes to this end state as early as practical, to classify wastes and to ensure that intermediate steps do not inhibit or complicate the achievement of the end state, and to accumulate assets for managing all wastes in the life cycle so that the accumulated liability at any stage of the life cycle is covered.

With regard to INPRO basic principle BP1 (waste minimization by design), a general comparison of a LWR system with a once through fuel cycle (OTFC) to a system of fast reactors (FR) with a closed nuclear fuel cycle (CNFC) shows significant advantages of the latter.

First of all, a FR can be operated at higher temperatures than a LWR resulting in a higher thermal efficiency, thus generating less waste per power (MW_e) produced.

Secondly, the radiotoxicity of nuclear waste, i.e. spent fuel to be put in final storage in a OTCFC system can be reduced significantly by recycling of plutonium and by partitioning and transmutation (P&T) of minor actinides (and specific fission products) in a CNFC system. However, it has to be stated that reprocessing of spent nuclear fuel produces several additional secondary waste forms that also need geological disposal.

To evaluate the influence of recycling of Pu and minor actinides on waste management the same six different scenarios for a nuclear energy system (NES) were considered (with a

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[16] Unconventional resources like phosphates and seawater are neglected in this evaluation. See also Section 3.3.

constant output of about 400 TWh_e per year produced by about 60 GW_e) as described in Section 4.5 (environment) above. The following analysis was performed:

- A neutronic calculation using the COSI computer code (Ref. [17]) to determine the amount (per TWh_e) of Pu, Am and Cm sent to waste, number and volume of spent fuel assemblies, volume of vitrified and compacted high level waste (HLW) sent to interim storage and final geological disposal (see Table 4.9).

Table 4.9. Results of COSI calculation of waste management parameters.

	Unit/ TWh_e	**NES No.1**	**NES No.2**	**NES No.3**	**NES No.4**	**NES No.5**	**NES No.6**		
							2015 - 2035	2035 - 2095	> 2095
Pu + Am + Cm sent to the waste	Kg	27.9	18.7	6.66	0.93	0.15	0.55	6.2	4.8
Spent fuel assembly	number	1.025	0.45	-	0.13	-	-	-	-
Spent fuel assembly	m^3	5	0.9	-	0.034	-	-	-	-
Vitrified waste canister	number	-	1.64	1.85	1.59	1.49	0.83	2.73	1.49
Vitrified waste sent to interim storage	m^3	-	0.29	0.32	0.28	0.26	0.15	0.48	0.26
Vitrified waste to final disposal	m^3	-	0.65	0.73	0.63	0.59	0.33	1.08	0.59

It is stated in the Joint Study that transuranium elements (Pu, Am, Cm) represent the bulk of radiotoxicity of HLW in a final waste depository but as they do not migrate they are not causing a significant health risk. Thus, they represent a very small potential to impact the environment.

The evaluation of the calculated parameters presented in Table 4.9 demonstrates that in comparison to an open or once through fuel cycle (NES No.1) mono recycling of Pu (as in NES No.2) reduces the amount of Pu/Am/Cm to be put in final disposal by a factor of 1.5, and multi recycling (as in NES No.3) by a factor of four.

The most promising results are achieved for NES No.5, i.e. in a FR system with full Pu and MA recycling. For example, the amount of Pu/Am/Cm to be put in final disposal is reduced by a factor of about 200 in comparison to the open fuel cycle of NES No.1. NES No.4 shows (non-surprisingly) an intermediate position between NES No.3 and No.5.

The analysis results of the dynamic NES (NES No.6) confirm the results achieved for the "steady state" scenarios (NES No.1 to No.5).

The following Figure 4.2 illustrates this reduction of radiotoxicity of nuclear waste disposed in a geological depository in comparison to direct disposal of spent fuel if actinides (Pu and MA) are eliminated from waste before final disposal. It depicts the ratio of radiotoxicity of nuclear waste to be disposed to the radiotoxicity of natural uranium ore as a function of time.

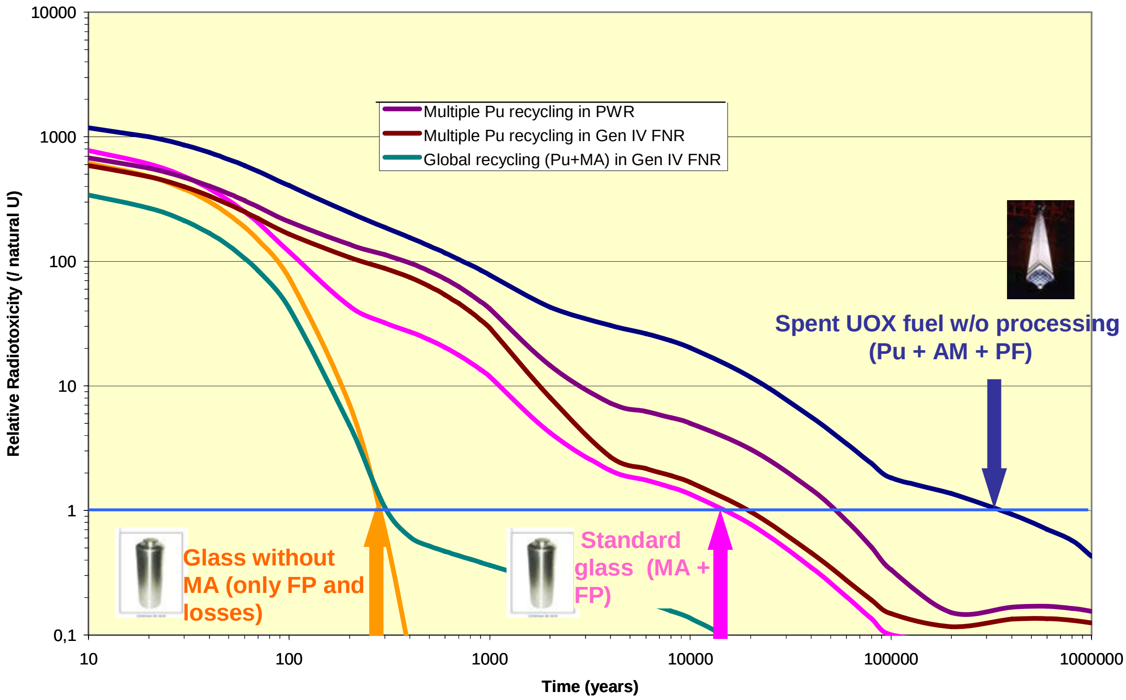

Figure 4.2. Potential radiotoxicity of nuclear waste disposed in geological disposal [16]. (AM = MA = minor actinides, PF =FP = fission products, FNR = fast nuclear reactor)

In an open fuel cycle of a LWR (NES No.1) operated with enriched uranium, spent nuclear fuel (i.e. the nuclear waste) put into a repository needs several 100000 years to reach an equivalent level of radiotoxicity as natural uranium ore. Multiple recycling of Pu in a closed fuel cycle of a PWR (NES No.3) already reduces this time of the corresponding waste to about 50000 years. Multiple recycling of Pu in a fast reactor system shortens this time further to about 20000 years. Finally, recycling of all actinides (plutonium and minor actinides) in a fast reactor system (NES No.5) reduces the radiotoxicity of the nuclear waste deposited dramatically (compared to an open fuel cycle of a LWR) reaching a level of radiotoxicity in the HLW equivalent to uranium ore after several 100 years.

Compared to a thermal reactor the fast neutron spectrum in a FR offers the potential to apply more efficiently the concept of P&T to eliminate MA in nuclear waste to be disposed. To illustrate this potential Table 4.10 presents a comparison of the mean effective cross sections of capture (σ_c)and fission (σ_f) of isotopes of uranium, plutonium and MA in a thermal neutron reactor (PWR with UO_2 fuel) and in a fast reactor. The higher the fission cross section of a MA is in comparison to the capture cross section the more such an actinide will be fissioned instead of transformed to another MA. As shown in Table 4.10 moving from a PWR spectrum (thermal or epithermal) to a fast neutron system reduces the capture to fission ratio of MA by a factor of up to 10, thus, resulting in a higher efficiency of transmutation of MA by fission.

Table 4.10. Comparison of mean effective cross sections of U, Pu and MA in a thermal and fast neutron spectrum [18]

Isotope	Thermal neutron spectrum (PWR)			Fast neutron spectrum (FR)		
	σ_f	σ_c	$\alpha = \sigma_c / \sigma_f$	σ_f	σ_c	$\alpha = \sigma_c / \sigma_f$
^{235}U	*38.8*	*8.7*	0.22	*1.98*	*0.57*	0.29
^{238}U	*0.103*	*0.86*	8.3	*0.04*	*0.30*	7.5
^{239}Pu	*102*	*58.7*	0.6	*1.86*	*0.56*	0.3
^{237}Np	*0.52*	*33*	**63**	*0.32*	*1.7*	**5.3**
^{241}Am	*1.1*	*110*	**100**	*0.27*	*2.0*	**7.4**
^{243}Am	*0.44*	*49*	**111**	*0.21*	*1.8*	**8.6**
^{242}Cm	*1.14*	*4.5*	3.9	*0.58*	*1.0*	1.7
^{244}Cm	*1.0*	*16*	**16**	*0.42*	*0.6*	**1.4**
^{245}Cm	*116*	*17*	0.15	*5.1*	*0.9*	0.18

In addition to the separation of Pu and MA from spent fuel also processes to separate and recycle certain fission products (FP) from spent fuel are investigated in the countries participating in the Joint Study, especially FP that are either generating high heat (with rather short half lives) or long lived (with low heat generation).

As said at the beginning of this section INPRO waste management basic principle BP1 asks for reduction of nuclear waste at the source. Summarizing the assessment in regard to BP1 of FR systems compared to thermal reactor systems the higher thermal efficiency of FR results in less HLW per unit power production (MW_e). Application of P&T results in less HLW with lower radiotoxicity to be geologically disposed. Comparing a thermal and a fast neutron reactor system, both with a CNFC, it was shown that it is possible to separate and transmute MA more efficiently in fast neutron systems. The main advantage of recycling of Pu and MA is the possibility to geologically store more HLW per volume of rock due to the reduced heat load. The issue of secondary waste production in reprocessing that also needs geological disposal was not treated in the Joint Study.

INPRO waste management basic principle BP2 asks for sufficient protection of human health and the environment. Both issues have been evaluated in the INPRO area of safety (Section 4.7) and environment (Section 4.5) and are not further treated here.

In regard to INPRO waste management basic principle BP3 (avoidance of burdens on future generations) the assessors claim that the necessary technology of waste immobilization will be available on an industrial scale on time to achieve a safe end state of nuclear waste of a CNFC-FR system.

The technologies for pre-disposal waste management including waste classification schemes of CNFC-FR systems required by INPRO basic principle BP4 (optimization of the waste process) are already existing.

Assessment of waste management of the Japanese FR development program

Additionally to the evaluation of INPRO area of waste management on the level of basic principles the Joint Study documented the evaluation of waste management on the level of

INPRO criteria of a nuclear energy system consisting of FR with closed fuel cycle based on the results of the Japanese feasibility study (FS, Phase II final evaluation) on a commercialized fast reactor cycle and the development program of fast reactor fuel cycle technology called FaCT. All INPRO criteria of all user requirements in the INPRO area of waste management were met by the program as laid out in the following.

Regarding INPRO user requirement UR1.1 (reduction of waste at the source) the assessor concluded that a system with FR and CNFC produces less waste than the LWR cycle system currently in operation. User requirements UR2.1 and UR2.2 (protection of human health and environment) are fulfilled as all national regulatory standards are met. Resources and sufficient time to develop necessary technologies are available and costs of the development are taken care of. Thus, UR3.1 and UR3.2 (end state of waste and inclusion of development cost) are met. The same classification of waste as for LWR will be used for FR and CNFC system and all intermediate steps of waste management are considered in FaCT, something that is requested by UR4.1 and UR4.2 (classification of waste and intermediate steps of waste management).

4.7. Assessment of CNFC-FR in the INPRO area of safety

INPRO has developed four basic principles (BP) in the area of nuclear safety based on the IAEA Fundamental Safety Principles (SF1), utility requirements such as EPRI Advanced Light Water Reactor Utility Requirements, and on an extrapolation of current trends assuming a large increase of nuclear power in the 21st century.

BP1 calls for an enhanced application of the concept of defence in depth (DID) with more independence of different levels of protection in the DID strategy. The corresponding user requirements (UR) provide recommendations how the designer/developer can achieve a higher safety level in new designs compared to a reference design (plants operating end of 2004 and well proven design) by intensified use of the DID concept in each of its five levels.

BP2 and the corresponding UR ask the designer — when appropriate — to consider the increased use of passive systems and inherent safety features in new designs compared to a reference design to eliminate or at least minimize hazards.

BP3 sets a high level goal by requesting the designer to reduce the risk level from nuclear facilities due to radiation exposure to workers and to the public so that this nuclear risk is comparable to the level of risks arising from facilities of other industries with a similar purpose.

BP4 and its user requirements ask for a sufficient level of R&D to be performed for new nuclear designs to bring the knowledge of plant characteristics and the capability of analytical tools to at least the same confidence level as for the reference design (operating end of 2004 and well proven design).

To illustrate some specific safety related differences of fast neutron reactor systems and thermal reactor systems the Joint Study produced the following Table 4.11.

Table 4.11. Comparison of safety related parameters of thermal and fast reactor systems

Parameter	Thermal reactor system.	Fast reactor system (CNFC-FR)
Core arrangement.	Near-optimal reactive geometry.	Not in its most reactive geometry.
Void coefficients.	Coolant void coefficient is negative.	Coolant void coefficient is positive or at-best zero (overall).
Stored energy in coolant.	High (PWR)	Insignificant
Chemical energy potential.	Low	High: sodium-water; sodium-air.
Inherent emergency heat removal capacity.	Low	High (particularly in pool type reactors).
Coolant activation.	Tritium and corrosion products.	Radioactive isotopes of sodium and corrosion products.
Burnup	~ 40 GWd/t	100 GWd/t (typical) – can be enhanced.
Pu content in irradiated fuel.	0.9 %	23.5 %
Total radioactive fission product inventory /GW_e.	X	X times the ratio of thermal efficiency of water reactor / fast reactor = about 30 % lower than thermal reactor.
Specific radioactive fission product inventory, immediately after shutdown.	3000 Ci/kg	6000 Ci/kg

In Table 4.11 above, compared to a thermal reactor system, some parameters listed demonstrate an improved safety level and others indicate a higher risk level of fast reactors.

Firstly, the disadvantages of fast reactor systems are addressed. Any change of the core arrangement of a thermal reactor, e.g. due to loss of coolant or core melting in case of a severe accident, will lead to a reduction of reactivity, whereas in a fast reactor such changes could lead to an increase of reactivity expressed for example by a positive coolant void coefficient. In case sodium gets in contact with water, the high chemical energy potential of the sodium water reaction results in sodium fires. The higher burnup of fast reactor fuel results in higher specific radioactivity of spent fuel of fast reactors compared to spent fuel of thermal reactors.

The disadvantages of fast reactor systems in regard to safety compared to thermal reactor systems are, however, compensated by several inherent features as shown in Table 4.11 above and by additional engineered safety measures. An example of inherent safety features are the excellent heat transfer characteristics and high boiling point of sodium that permits to design the reactor coolant system with a very low pressure, resulting in low stored energy of the coolant fluid. An example of engineered safety features is the installation of double walled pipes and vessels to avoid sodium fires.

Within the Joint Study China, France, India, Japan, and the Russian federation performed an INPRO assessment of the safety of a *fast reactor design*. However, each country looked at its national design that doesn't concur with the chosen reference model of a fast reactor presented in Section 4.1. It was noted that the existing experience with the operation of FR confirms that workers in such units receive low doses. One reason for the low radiation doses is the blind handling of spent nuclear fuel in a sodium cooled reactor, i.e. spent fuel is always covered or shielded by sodium and not visible to the operator.

China looked at their conceptual design of the CFR-1000. This design is characterized by a pool type concept, use of sodium as coolant, provision of small reactivity margins, core with low power density, decay heat removal by natural convection, and a passive shut down system. A qualitative assessment of INPRO safety user requirement UR1.1 (increased robustness of design) was performed and agreement confirmed.

France assessed the safety of new components of their national nuclear energy system, i.e. the next generation of PWR to be installed, the EPR (Generation-III+) as well as their designs of fast reactors — sodium or gas cooled — under development. The design of the EPR was found to fully comply with practically all INPRO requirements in the area of safety. The French concept of the fast sodium cooled reactor (SFR) is characterized by an improved (compared to an existing design like the European Fast Reactor) core design with a decreased sodium void coefficient. Also the shut down and decay heat removal systems have been further developed and a core catcher integrated into the design to retain molten core material after a hypothetical severe accident. The assessor concluded that the safety of their fast reactors will be at least comparable to the EPR design.

India performed an assessment of safety of their fast reactor concept, however, no details of the results are provided in the Joint Study.

A complete assessment of safety of the *Japanese* sodium cooled fast reactor (JSFR) using the INPRO methodology (at the criterion level) was performed. This detailed assessment confirmed that the JSFR design does (or is expected to) fulfil all INPRO criteria in this area. The assessor concluded that the design of JSFR is sufficiently robust, i.e. simple and with enough design margins and that by the inclusion of passive and inherent safety features into the JSFR design an adequate safety level is achieved.

By the *Russian Federation* the BN-800 fast reactor design was assessed in regard to safety. The frequency of a major release of radioactivity into the environment was calculated to be below 10^{-7} per reactor year, avoiding a need of evacuation after a severe accident. Thus, several criteria of INPRO safety user requirement UR1.5 were met.

Safety of *fuel cycle facilities* of fast neutron reactor systems was addressed within the Joint Study by *India* primarily. A detailed description of all safety issues related to the operation of facilities of the front end and back end of a closed nuclear fuel cycle is provided in the Joint Study (for details see Volume 9 on Ref. [2]) but no specific results of an INPRO assessment of safety of such facilities are presented. The main safety issues in fuel cycle facilities are avoidance of criticality of fuel material, and release of toxic chemicals together with radioactive material. The higher Pu content and burnup of fast reactor fuel compared to thermal reactor fuel enhances the risk and possible consequences of accidents in fuel cycle facilities.

CHAPTER 5
EXAMPLES OF RESEARCH AND DEVELOPMENT FOR CNFC-FR

CNFC-FR systems must be economically attractive, safe, environmentally benign and proliferation resistant to become viable alternatives to the conventional sources of energy. The Joint Study concluded that a comprehensive R&D program with an interdisciplinary approach and international collaboration wherever possible is essential to achieve the defined objectives of CNFC-FR systems under development.

To achieve competitiveness in the area of economics primarily the capital costs of fast reactor systems are to be reduced. Possible measures are a reduction of construction time and of number and size of components, and extending the life of the components. Figure 5.1 is an example of the planned measures to reduce the capital costs of the fast reactor.

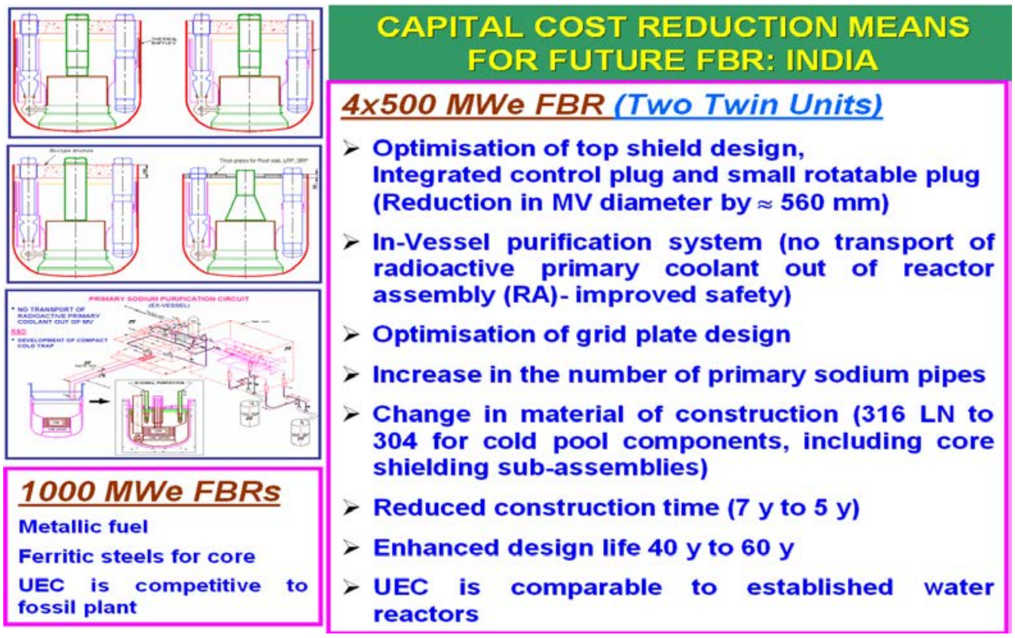

Figure 5.1. R&D to reduce capital cost of fast reactors in India.

An example of a reduction of size is the fast reactor under development in Japan that in comparison to the prototype reactor MONJU is expected to generate five times more power within a quarter of the needed site area. This reduction of size results in a significant decrease of the total quantity of concrete and steel used for construction per MW_e installed; concrete and steel are main contributors to the capital costs.

Another example of successful R&D efforts is the significant reduction of specific capital costs to about 1200 $/kW of the Russian fast reactor BN-800 (under construction) in comparison to the prototype BN-350 with about 1900 $/kW (see Figure 5.2).

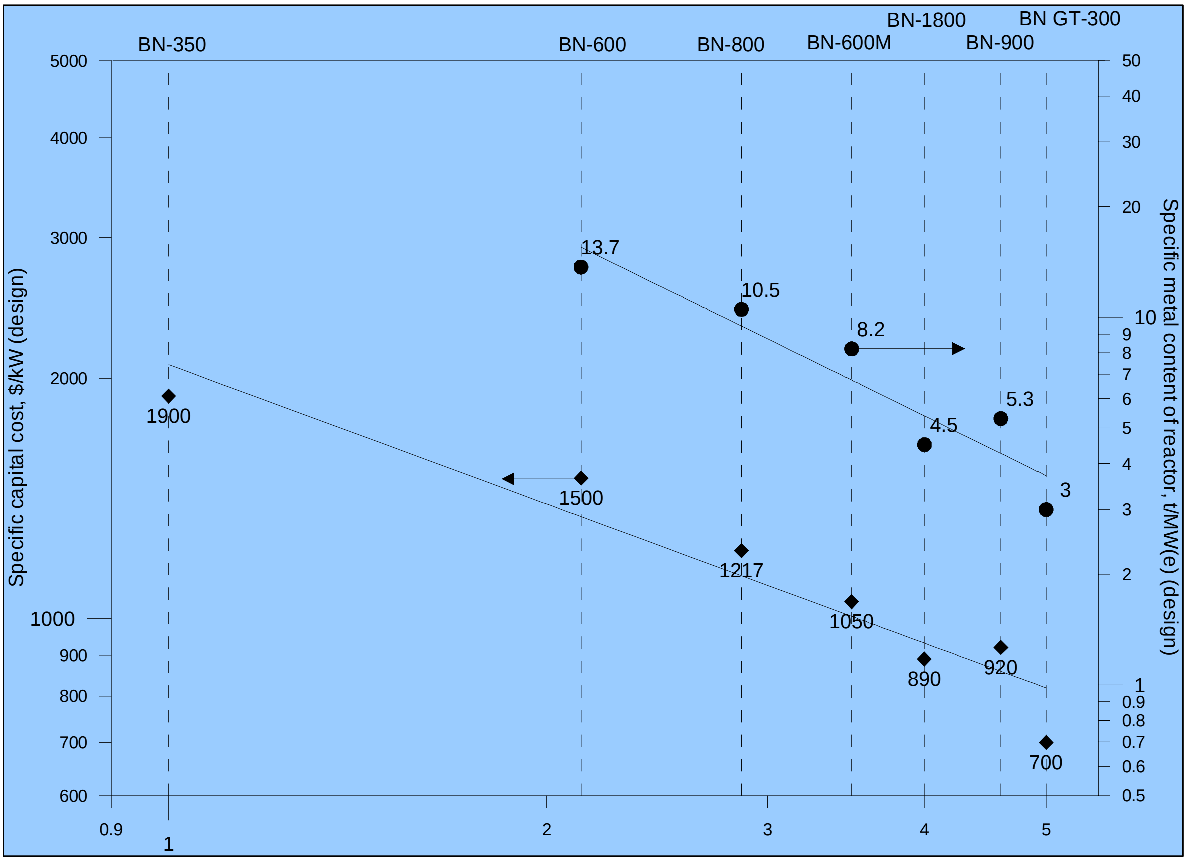

Figure 5.2. Reduction of capital cost of fast reactors due to R&D in Russia.

Extending the life of components requires development of advanced materials, and manufacturing and maintenance technologies. To be able to study radiation damage in advanced materials suitable test facilities such as accelerators are needed that could be operated on an international basis.

To increase the safety level of fast reactors by reducing hazards of radiation exposure to workers advanced shielding materials such as boride/rare earth compounds are under development. Other safety related measures considered are the use of hard facing materials that do not get activated and in vessel purification systems of sodium. To prevent reactor accidents sodium pumps with high inertia, and automatic negative reactivity insertion devices are investigated. To enhance safety after severe accidents measures to prevent recriticality in molten core configurations are studied.

Reduction of public exposure could be achieved by advanced cover gas purification technologies that minimize the release of gaseous fission products of fast reactors.

Increasing the robustness of important components in reprocessing and waste management facilities is key to enhancing the economics of fuel cycle operations. Advanced materials are under development that reduces corrosion of components used for processing of spent fuel such as dissolver, evaporator and waste tanks thereby enhancing the life of these equipments.

To improve the economics of equipment in nuclear fuel cycle facilities (e.g., dissolvers, evaporators) R&D is required to reduce the size of process equipment without sacrificing criticality margins. Advanced condition monitoring systems are to be developed that would provide continuous information regarding the condition of these equipments.

To increase the proliferation resistance, R&D of reprocessing technology is ensuring recovery of uranium and plutonium without separation.

To simplify waste management and reduce the demand for repository space reprocesses are to be developed for recovery of minor actinides and long lived or heat producing fission products. The development of ceramic matrices with long term stability and higher capacity for waste loading is another important area of R&D for final repository.

The Joint Study concluded that it is possible to identify generic areas for international collaboration such as development and testing of materials, in service inspection technologies, modelling and validation of codes, probabilistic methods for safety analysis of fuel cycle facilities.

Within the frame work of INPRO the following Collaborative Projects have been initiated by participants of the Joint Study:

- Integrated Approach for the design of safety grade decay heat removal system for LMR (called **DHR**);
- Assessment of advanced and innovative nuclear fuel cycles within large scale NES based on CNFC concept to satisfy principles of sustainability in the 21st century (called **FINITE**);
- Investigation of technological challenges related to the removal of heat by liquid metal and molten salt coolants from reactor cores operating at high temperatures (called **COOL**);
- A Global Architecture of NES based on thermal and fast reactors including a closed fuel cycle (called **GAINS**).

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ANNEX
FEEDBACK ON INPRO METHODOLOGY

In this chapter the feedback of the Joint Study on the INPRO methodology (Refs [1] and [2]) is presented starting with general recommendations and then providing specific comments for each area of the methodology.

General feedback on the INPRO methodology

The Joint Study concluded that the INPRO methodology for two different designs under development that both fulfil all INPRO criteria cannot define clearly which one is the better option. Thus, it asks for an increased capacity of discrimination between several options of designs to be developed to select the optimum choice. One possible approach to follow this recommendation could be to introduce scaling for INPRO numerical criteria and taking into account the margin between the value of the indicator and the corresponding acceptance limit.

The INPRO methodology should include a clear description how to handle a situation during a development project where further improvement of one INPRO indicator could lead to a degradation of another indicator, i.e. how to achieve a balanced design either for a specific nuclear energy system (NES) component or the complete NES.

Another general recommendation is to enlarge the capabilities of INPRO methodology to define R&D goals beyond the existing INPRO acceptance limits.

The absence of specific INPRO Criteria for some nuclear fuel cycle facilities was mentioned and it was recommended to create such criteria.

An online data base should be created that includes all necessary information of currently available reactors and nuclear fuel cycle facilities to be used in an INPRO assessment.

Feedback on the area of economics (Volume 2 of Ref. [2])

The discussion of the investments required to develop an NES from the pre-conceptual stage to the commercially proven stage, presented in the INPRO Manual should be expanded and criteria related to investments required for development, as distinct from those related to deployment, should be developed.

Another related issue that could also be discussed further in the Manual, is the discount rate to be used when comparing costs of energy alternatives, especially when considering the long time frames over which an NES is expected to operate, as well as the detailed models that are to be used in calculating costs, e.g., the Merchant Cash Flow model used by MIT (Ref. [19]).

Feedback on the area of infrastructure (Volume 3 of Ref. [2])

The Joint Study stressed the need to develop a method how to quantify the overall added value of a proposed nuclear installation that should compensate the necessary investment in infrastructure to support a nuclear installation.

The difficulty was mentioned to assess the infrastructure of a system that is in early stage of development or deployment due to the lack of available data.

It was noted that countries with little or no nuclear experience would benefit if the INPRO Manual provided an example of an assessment for such a country and for a country with

nuclear power experience. In this connection, it was suggested that countries with nuclear experience might assist in-experienced countries, in performing an assessment in the area of infrastructure with the assistance of the IAEA (Department of Technical Co-operation and Nuclear Power Engineering Section of the Department of Nuclear Energy.)

Feedback on the area of waste management (Volume 4 of Ref. [2])

Many of the suggestions seem to relate to a need to improve the clarity of the INPRO Manual. One suggestion for doing so is to utilize evaluation parameters as is done in the INPRO area of infrastructure. Another related suggestion is to specify the waste management system for a reference nuclear energy system, such as a once through LWR system, and assess this system. This would necessarily provide a reference set of values for indicators and acceptance limit and a given NES could then be compared with the reference INS.

The INPRO criteria should be adjusted depending on the stage of development of an INS. The existing criteria used in waste management represent what should be achieved when an is fully developed. At an early stage of development it may not be possible to determine whether a given criterion will be met. In such a circumstance the assessment would be incomplete and the criterion in question would have to be re-evaluated at a later stage of development. On the other hand if, at an early stage of development, the judgment is that the criterion might not be met then this result would indicate the need to modify the development plan to address this potential shortcoming of the INS.

Feedback on the area of proliferation resistance (Volume 5 of Ref. [2])

The Joint Study expressed a need for clarification how to deal with lack of existing data in case of early design stages.

An internationally accepted standard of proliferation resistance (PR) of a nuclear system (or component thereof) and a clear description in the INPRO manual of a standardized method how to evaluate the level of PR is needed.

Feedback on the area of environment (Volume 7 of Ref. [2])

A need was expressed to clarify in more detail in the INPRO Manual how to apply the methodology in this area.

It was claimed that the INPRO methodology in this area does not include sufficient capability to demonstrate the benefits of introducing partitioning and transmutation (P&T) technologies that have a potential to reduce environmental impacts of nuclear waste.

Another suggestion is to reference, in the INPRO Manual, the “Basic Environmental Law” issued by the United Nations in 1993.

A general recommendation was made to extend the methodology in this area to cover accident situations.

Feedback on the area of safety (Volume 8 and 9 of Ref. [2])

A need was defined to further develop the INPRO methodology for application on nuclear fuel cycle facilities, particularly, in the area of safety. In the current version of Volume 9 of the INPRO manual no explicit guidance is provided on how to treat storage facilities for spent nuclear fuel.

An overlap of requirements, especially, on the level of criteria, i.e. frequencies of occurrences, consequences of events was noted between INPRO safety basic principle BP1 (enhanced defence in depth) and BP2 (inherent safety characteristics and passive systems).

A commonly agreed assessment of an existing nuclear energy system seems necessary to be used as a benchmark (or reference) case; a future NES could be compared with such a benchmark and its improved level of safety demonstrated.

The documentation of the INPRO methodology in the area of safety needs more explanation of technical terms used in the Manual (Volume 8 and 9 of Ref. [2]).

Similar to other INPRO areas it is requested to clarify how to treat early design stages with a lack of data.

It was further suggested that in case for a design under development a large (excessive) design margin was found for a given INPRO parameter, one might consider whether some trade offs could be considered to balance the overall design.

LIST OF ABBREVIATIONS

AECL	Atomic Energy of Canada Limited
ADS	accelerator driven system
AGR	advanced gas reactor
AHWR	advanced heavy water reactor
ALARP	as low as reasonably practical, social and economic factors taken into account
BP	basic principle (INPRO)
BR	breeding ratio
CDA	core disruptive accident
CEA	Atomic Energy Commission (France)
CEFR	China Experimental Fast Reactor
CIAE	China Institute of Atomic Energy
CNFC-FR	closed nuclear fuel cycle with fast reactors
CR	criterion (INPRO)
CP	collaborative project (INPRO)
DBE	design basis earthquake
DU	depleted uranium
EFR	European fast reactor
EPR	European pressurized reactor
FC	fuel cycle
FCF	fuel cycle facility
FOAK	first-of-a-kind
FP	fission products
FBR	fast breeder reactor
FR	fast reactor
FSA	fuel subassembly
GCC	gas combined cycle
GFR	gas fast reactor
GHG	green house gas
GIF	Generation IV International Forum
HEU	highly enriched uranium
HLW	high level waste
HM	heavy metal
HTGR	high temperature gas reactor

HTR	high temperature reactor
HWR	heavy water reactor
I&C	instrumentation and control
IGCAR	Indira Gandhi Centre for Atomic Research (India)
IIASA	International Institute for Applied System Analysis
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
INS	innovative nuclear energy system (INPRO)
IPCC	Intergovernmental Panel on Climate Change
IPPE	Institute for Physics and Power Engineering (Russian Federation)
IRR	internal rate of return
ISI	in-service inspection
JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
LCA	life cycle assessment
LEU	low enriched uranium
LWR	light water reactor
MA	minor actinides
MFA	material flow assessment
MNFC	multilateral nuclear fuel cycle
MOX fuel	mixed oxide uranium-plutonium fuel
MS	Member State (IAEA)
NNEGC	National Nuclear Energy Generating Company (Ukraine)
NPP	nuclear power plant
NPT	Non-Proliferation Treaty
NOAK	N^{th} of a kind
O&M	operation and maintenance
OTFC	one-through (open) fuel cycle
P&T	partitioning and transmutation
PHWR	pressurized heavy water reactor
PR	proliferation resistance (INPRO)
PSA	probabilistic safety analysis (assessment)
PUREX	plutonium and uranium recovery by extraction
PWR	pressurized water reactor
RBMK	graphite moderated light water cooled fuel channel reactor
R&D (RD&D)	research and development (research, development and demonstration)

ROI	return on investment
SF (SNF)	spent fuel (spent nuclear fuel)
SFR	sodium fast reactor
SRES	special report on emission scenarios
SWU	separation work units
TRU	transuranic elements
VVER (WWER)	water cooled, water moderated power reactor

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