POSTER PRESENTATIONS

NUCLEAR DESALINATION: A POSSIBLE SOLUTION TO LATIN AMERICAN REGION'S ONGOING WATER SCARCITY

S.M.G. DE SOLER, L.B. RAMILO, N.R. COPPARI, R. CORCUERA

National Atomic Energy Commission
Dept. of Nuclear Reactors and Power Plants
Argentina

Abstract. Argentina has a desalination project, with support of the International Atomic Energy Agency (IAEA) as a Coordinated Research Project (CRP): Economic Research on, and Assessment of, Selected Nuclear Desalination Projects and Case Studies in Argentina and Latin America. This project's objective is to acquire the capability to evaluate the feasibility of certain nuclear desalination options as well as possible sites in Argentina, or in other areas of Latin America. On the other hand, Argentina is also developing a small nuclear power plant, the CAREM nuclear power plant, which can be used as an energy source for water desalination. In Argentina's desalination project, the CAREM nuclear power plant will be adopted as the energy source. A combined cycle gas turbine power plant will be used with comparative purposes. This paper shows the studies carried out up to the present in this desalination project.

1. INTRODUCTION

It is clear, from a global perspective, that as the world's population growth and industrialization continues there will indeed be a scarcity of fresh water resources. By 2025, 1,8 billion people will live in countries or regions with absolute water scarcity. Of all the globe's water, 97 percent is salt water from the oceans.

Technologies of water-desalination -available today in the world- require energy supply, such as heat or electrical energy. Then, the use of nuclear energy for electricity and potable water production is attractive. Also, this option is technically feasible and a safe alternative to fossil energy options.

Argentina has a desalination project, with support of the International Atomic Energy Agency (IAEA) as a Coordinated Research Project (CRP): Economic Research on, and Assessment of, Selected Nuclear Desalination Projects and Case Studies in Argentina and Latin America. This project's objective is to acquire the capability to evaluate the feasibility of certain nuclear desalination options as well as possible sites in Argentina, or in other areas of Latin America. Argentina is also developing a small nuclear power plant, the CAREM nuclear power plant, which will be adopted as the energy source in this project. A combined cycle gas turbine power plant will be used with comparative purposes.

This paper shows the studies carried out up to now in this desalination project. At first, the evaluation's results of the Argentinian regions and the regions of the rest of Latin America are presented, in particular where the shortage of fresh water represents an important restriction for its socioeconomic development. The conclusions that were obtained in the study of the available desalination technologies, in particular the advantages and disadvantages of each of them and their main associate costs, are also included. As regards the energy supply the main characteristics of the nuclear power plant, CAREM, are also described. Finally, the main characteristics of the selected site are indicated, the reasons that were kept in mind for the selection of the desalination technology are included, and the reasons why the nuclear desalination of seawater could represent a solution to shortage of fresh water in the selected site are delineated.

2. LATIN-AMERICAN REGIONS ONGOING WATER SCARCITY

A study for different countries of Latin America has been carried out, in particular where the shortage of fresh water represents an important restriction for its socioeconomic development.

At first, the characteristics of each country were studied. This study was focused on the coastal areas, which have scarcity of fresh water and have seawater as well. It was considered that they would be the regions with more potentiality to install desalination plants. In order to achieve it, the study and analysis of relevant literature search has been carried out. It focuses on issues of the United Nations Economic Commission for Latin America and the Caribbean [1-6]. A report about water resources in Latin America is also available [7] as well as relevant general literature [8 to 10], and several particular publications from other countries [11-14].

Furthermore, a specific study for different Argentinian Provinces has been performed. For these studies, bibliography available in the country –elaborated at a national level or by each province in particular- was used [15-29].

These studies have been conducted for regions with economic importance, potential capacity of development and big surfaces with restrictions of fresh water and availability of seawater. The availability of energy supply was also considered.

As a result of that first evaluation, four regions were selected. One of them is in Argentina and the others are in rest of Latin America. Figure 1 show the location of these Latin-American's regions.

FIG. 1. Latin-American regions with ongoing water scarcity.

2.1. Argentina

According these studies, the Argentinian region is on the central coast of the Argentine Patagonia, which includes the coast of Rio Negro Province, the coast of Chubut Province and the northern coast of Santa Cruz Province.

The representative sites in this region are four cities:

— San Antonio Oeste, Río Negro Province.
— Puerto Madryn, Chubut Province.
— Comodoro Rivadavia, Chubut Province.
— Puerto Deseado, Santa Cruz Province.

The study of data corresponding to these cities has proven that [15, 16, 26 - 28, 30 - 34)

— There is no aquifer with fresh water.
— The rainfalls are insufficient.
— They do not have surface fresh water (rivers, lakes).
— The fresh water is transported from big distances.
— The climate is called "Arid Patagonic".
— The supply of electric energy is made from the Patagonic Interconnected System (regional), with a 132 kW electric transmission line. The offer is of 774MW; 2/3 comes from hydro power plants and 1/3 of thermal power plants.

This study has proven that:

— **San Antonio Oeste** has severe scarcity of water, but few possibilities of industrial growth [35, 36].
— **Puerto Madryn** has a severe scarcity of water and many possibilities of industrial growth [36 - 41]. Although this city has the highest number of inhabitants and a growing heavy industry (Aluar), it is important to admit that the community makes strong pressure in favor of renewable energy. The success of the project strongly depends on public acceptance, fact that has already been seen in different experiences in the world.
— **Comodoro Rivadavia** has, at present, solved the scarcity of water. It has an aqueduct, and water delivery has recently started [42]. The cost of this solution would not be the best.
— **Puerto Deseado** has a severe scarcity of water and many possibilities of industrial growth [36, 43 – 49]. Regarding the energy supply, Puerto Deseado is connected to the Patagonic Interconnected System with an electric line of 132 kW from Pico Truncado to Puerto Deseado. This city is situated in the end of this line. In order to increase the reliability of supply, it would be very important that the city should have a power plant [32 – 34]. On the other hand, it is significant to emphasize that Puerto Deseado is located in the province of Santa Cruz, whose main activities are petroleum oil production, which represent 25% of the total crude output of the country, fishing -which is growing- and sheep raising. The industrial activity of the province concentrates on fishing processing plants and cold-storage activity. Other important industries are commerce and tourism, which provide a significant source of employment. The fiscal outlook of the province is quite good, since it presents a well-balanced public expenditure structure, setting aside 20% of the total expenses to public works and infrastructure investment [50]. Finally, it is worth pointing out that Santa Cruz social indicators are very good, thanks to its low unemployment rate.

2.2. Rest of Latin America

According these studies, three other regions of Latin American were selected, they are:

— One region is formed by the numerous islands of the Caribbean. This region includes an area that expands forming an arch from Florida in the United States of America and the Yucatan Peninsula in Mexico, until the northeastern coast of Venezuela. The biggest islands are in the northwestern part of the region and they represent more than 80% of the total surface. The Lesser Antilles include the East and Southeast of the region, from Puerto Rico Island to the Venezuela Gulf. Many of the islands are of volcanic origin and in some of them their lands have drainage problems. Rains concentrate in summer months and there are serious restrictions of fresh water, not only in amount but in quality [1, 9, 10, 51]. Today, there are already plants of seawater desalination as source of fresh water provision on several islands. E.g.:
  - In Antigua, there are five reverse osmosis units which provide water to the Antigua Public Utilities Authority, Water Division [8].
  - In the British Virgin Islands, all the water used on the island of Tortola, and approximately 90% of the water used on the island of Virgin Gorda, is supplied by desalination [8].
  - Bahamas, Curaçao, Aruba, Bonaire and the U.S. Virgin Islands have plants of seawater desalination, too [9].

— Another region is the coastal area of Venezuela, from the Esparta Islands in the east until the Unare Valley in the west. Their center is in the cities of Cumaná and Barcelona. In this region, the Cariaco Gulf has the minimum rainfall per year. Also, in the coastal area of the Cariaco Gulf, the Araya Peninsula and Margarita Island the lands are frequently of saline type [2].

— The last region corresponds to the Dry-Pacific hydrographic system. It is a long strip and it narrows on the western coast of South America, between parallels 3 ºS and 31 ºS. Its eastern limit is the Andes division of waters and its western limit is the Pacific Ocean. Toward the north, it extends until the Guayaquil Gulf, in the center it includes the whole western slope of the Peruvian mountain and in the south it includes the so-called "Small North" and "Big North" of Chile. It is the most arid region in Latin America and it is characterized by its coastal marginal deserts. The natural flows of water are a few and seasonal. In summer they can be of alluvial type. Special atmospheric conditions occur along the arid coast of Chile and southern Peru, where the clouds settle on the Andean slopes and produce what is locally known as "camanchacas" (thick fog). This is characteristic of a narrow strip, which is not more than 40-50 km wide. In the whole region the arid climate prevails. In this climate -in the coastal area- the desert and the steppe climates can be distinguished, with virtually no rainfall, such as in Iquique, with an average of annual rainfalls of 2 mm and Lima with 25 mm [1, 2]. Hence, alternative sources of freshwater are required. In this region there are antecedents of plants in operation and of projects of installation of seawater desalination plants, such as:
  - Antofagasta, Chile, where there is a project of a plant for seawater desalination, by reverse osmosis process [13].
  - Ilo, Peru, where there is a small plant of seawater desalination, by multi-flash steam process [14].

3. DESALINATION TECHNOLOGIES

In order to know the advantages and disadvantages of each seawater or brackish water desalination technology, and the distinctive characteristics of each of them, that make them better adapted to different uses and site conditions, main water desalination technologies have been studied.

Today a number of technologies are used to remove salt from seawater in the world. These technologies are basically divided in two general categories: thermal processes and membrane processes.

The three major thermal processes that are being used are: multistage flash distillation (MFS), multiple effect distillation (MED), and vapor compression distillation (VC).

Membrane technology used for seawater treatment is the reverse osmosis (RO). Electrodialysis is a membrane process commonly used in desalination of brackish water.

3.1. Several technical advantages - disadvantages, improvements and innovations of each technology [52 – 63]

The thermal processes, MFS, MED and VC technologies, are generally excellent to remove dissolved mineral from water. These technologies produce high–quality water, in some cases having less than 10ppm TDS [63] (Total Dissolved Solids). In these processes the resulting Langelier index (LSI) of about -8.1 [53] is extremely corrosive, and chemicals (usually lime and caustic) are added before fresh water distribution. The thermal processes also require some form of chemical (additive or acid) pretreatment to minimize the formation of scale (mineral deposits), but these plants do not generally require extensive prefiltration. Screens at the seawater inlet with a basket type strainer are sufficient.

MSF distillation is a proved and mature technology. This technology is advantageous in large capacity ranges where thermal energy in the form of low pressure steam is available. In dual purpose plants, where projects have specified both power and water, using oil-fired boilers for power generation, MSF was chosen. Seawater requires, for this process, chlorination for biological control and some chemical pretreatment to minimize the formation of scale. High temperature distillation plants can use either a high temperature additive or an acid one or a combination of both. In the case of low temperature, polyphosphate is suitable. The recovery rate of this process is dependent upon feedwater quality and operating temperature. Its maximum recovery rate, that is about 12-20% [53], is much lower than that of the other processes. This technology has high-energy consumption. In the MSF process the brine recirculation configuration is the most used arrangement. The brine recirculation flow rate is about nine times the production flow. The brine recirculation pumps account for a large part of auxiliary power consumption of about 3.5 kWh/m^3 [54]. Long-tube plants can readily be designed with a large number of stages, giving high GOR (Gain Output Ratio), but with reduced pumping power. With modern materials there is also the possibility of increasing a maximum operating temperature and reintroducing the “once-through” plant instead of brine recirculation. The maximum temperature increase can be accomplished by removing scaling components and/or developing better scale inhibitors.

MED distillation can achieve high heat transfer rates due to the film boiling and condensing conditions. The major advantage of the MED process is the ability to produce a significantly higher GOR than MSF for a given source steam temperature. The number of effects of a MED plant is much lower than those in an equivalent MSF plant. Power consumption of MED plants is lower than that of MSF, around 2.0 kWh/m^3, because there is no requirement to recirculate large quantities of brine. Its maximum recovery rate, that is about 30-40% [53], is also higher than that of the MSF plants. Seawater for this process requires chlorination for biological control and some chemical treatment to minimize the formation of scale. High temperature MED plants can use either a high temperature additive or an acid one or a combination of both. It is suitable in the case of low temperature polyphosphate. The low-temperature MED claims minimal power consumption and reduction of scaling and corrosion problems, due to low temperature operation (70°C), which allows use of cheaper construction material and minimal requirements for intake and pretreatment systems. A new configuration based on plates for heat transfer surfaces improves efficiency, economics and simplified operation in MED plants. The performance of MED plants can be improved still further by means of vapor compression (MED/CV), whereby part of the vapor formed in a low – temperature effect is recompressed and reintroduced to the first effect. The thermal vapor compression (TCV) is the method in large plants.

TVC technology uses live steam from a boiler to suck in the vapor from the compartment and recompress it. The reliability of the installation is improved because there are no rotating parts apart from low pressure pumps. It gets over the scaling problem because it is a low temperature system. At present, this taking impulses the development of ejectocompression, which is a combination of thermal compression and multiple effect (MED/TCV mentioned in a previous paragraph). In order to achieve a good performance from a thermocompressor, a substantial amount of medium pressure steam, higher than 1 MPa is normally required. The other VC process is the MVC process. MVC is a thermodynamically efficient process. The thermodynamic efficiency of the MVC is derived from the application of the "heat pump" principle. The latent heat required by the system is continuously recycled by a large volumetric flow compressor, thus eliminating the need for cooling water for heat rejection as in MSF or MED processes. This technology has a limited production level from 4,000 to 5,000 m^3/d, because of the size of the compressor that is available in the market. The maximum recovery rate of the VC plants, that is about 40-50% [53], is also higher than that of the MSF and the MED plants. Power consumption of single purpose VC plants is lower than MSF and MED plants.

The membrane technology used for seawater treatment, RO, is an effective method for removal of dissolved solids, organic contaminants, bacteria and viruses. The water quality produced with it is typically 410-500ppm [63] TDS. This TDS is predominantly sodium chloride and it is within the World Health Organization drinking standards (less 500ppm TDS), but it is not sufficiently pure for industrial use. When high quality water is required, the permeate must be treated a second time using a BRWO (Brackish Reverse Osmosis) system in what it is normally called a permeate two-pass process [63]. In this process the resulting Langelier index (LSI) is negative [53], and it is corrosive to distribution piping. Chemicals such as lime are added to increase the LSI to a positive level, before fresh water distribution.

In recent years RO has become a more attractive technology for seawater desalination because of lower membrane costs and its lower energy requirements compared to distillation technologies [54, 59]. RO systems are modular size, so any plant capacity can be obtained by installing the proper number of units. There is no optimum number unit size for RO. Limited extensions can be made by adding additional membrane stacks and pumps. In case RO is fully implemented, storage capacity to overcome maintenance periods is no longer necessary. Its major advantages are high recovery rate (40% with tendency to 60% [62,64]), high availability (90-95% [53, 59]), low energy consumption [53, 54], simple operation, little process supervision, operation at low temperatures and associated significant use of plastic and low corrosion materials, flexible production ratio of power to water, modular construction. RO plants require significant prefiltration in terms of large sand/media filters followed by cartridge filters (mesh size 5mm or smaller) process. The extent of the prefiltration may be reduced by the use of beach wells instead of open seawater intakes. RO plants chemical pretreatment can be made using acid, a scale inhibitor or a combination of both. RO may require coagulants if the seawater is turbid, and usually chlorination and dechlorination if there is biological fouling potential. Pretreatment equipment is a key item in any RO plant. The ultrafiltration (UF) technique promises to be a superior alternative to conventional prefiltration, providing a filtrate quality that is free of suspended solids and micro-organisms. Most seawater RO desalination systems used today are confined to approximately 40% [64] conversion of the feed water, since most of commercially available RO membranes do not allow high pressure operation of more than 7 MPa [64]. To increase the pure water recovery by a membrane module from the conventional 40% to 60% is a trend observable in sea water RO technology. Since the osmotic pressure increases when the recovery increases from 40% to 60%, the development of a high pressure vessel, as well as the development of a membrane that will show little compaction under high pressure, is necessary. There are recent reports about the development of a reverse osmosis membrane that was suitable for operation at 9 MPa [62,64]. Increasing the temperature of the feeding water can also improve the recovery of the plant [52].

Another method of reducing the overall cost of desalting can be the use of hybrid systems. A hybrid system is a treatment configuration made up of two or more desalination processes[52,65). Experimental testing of MSF/RO hybrid systems has demonstrated significant gains in product water recovery and reductions in electric energy consumption [59]. On the other hand, the main associated cost of each desalination technology has been collected [51, 54, 58 - 60, 66 – 72].

A summary of operating requirements for three processes: MSF, MED and RO-is included in Table I. Complementary information, a summary of the data in relation to the investments and operation costs [51, 54, 58, 59, 67 - 70] for the three mentioned methods is included in Table II.

4. CAREM NUCLEAR POWER PLANT

The main characteristics of the nuclear power plant, CAREM [73], are described in this section, in relation with the energy supply for the site and for the desalination plant.

CAREM power plant has a potential use for non-electric applications such as nuclear desalination. In this application this nuclear power plant could be used either as heat source or electrical energy supply.

The CAREM design is based on an integrated light water reactor with slightly enriched uranium fuel. It is an indirect cycle reactor with some distinctive features that greatly simplify the design and also contribute to a high level of safety. The main design characteristics are:

— Integrated primary cooling system
— Primary cooling by natural circulation for lower power module and assisted circulation for high power.
— Self-pressurized
— Passive safety systems

The primary cooling system is integrated. The reactor pressure vessel (RPV) includes the core, the steam generators, the whole primary coolant and the absorber rod drive mechanisms.

The fuel assembly components are similar to those of a conventional PWR design.

Core reactivity is controlled by Gd_2O_3 as burnable poison in specific fuel rods and movable silver–indium-cadmium absorber rods. The control rods drives are hydraulic and are placed inside the RPV. Chemical compounds are not used for reactivity control during normal operation. In Figure 2 an outline of the primary system for module of lower power is shown.

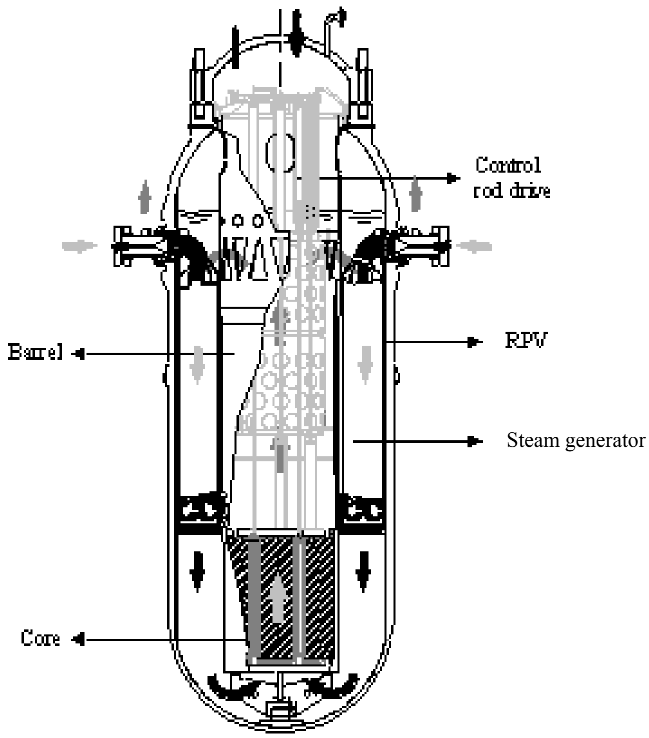

FIG. 2. Reactor Pressure Vessel – Primary system.

Table I. Operating requirements for three processes: MSF, MED and RO

OPERATING REQUIREMENTS	DESALINATION PROCESS		
	MSF	MED	RO
Energy:	-----	-----	70% of total production cost [68]
Heat (MJ/m^3)	161.5 [54]; 230 [68]	146.2 [54]; 209 [68]	0.000 [54; 59]
Steam consumption (t/h)	24t/h [51]	27t/h(a) [51]	-----
Auxiliary power (MJ/m^3)(b)	24.9 [54], 12.06 [51], 18 [68]	15.9 [54]; 7.92 [51]; 10.8 [68]	29.1 [54]; 50.4 [51]; 43.2 [68]
Auxiliary power (MJ/m^3)(c)	-----	-----	24.2 [54]; 23.4(d)); [59]; 19.08(e); [59] 39.6 [51]; 14.4 [68]; 22 [70]
Operation & maintenance:	Maintenance: 2% of investment cost [51].	Maintenance: 2% of investment cost [51].	Maintenance: 2% of investment cost [51]; 23% of total production cost (with membranes and other parts included) [68]; 3% of investment cost [70]. Maintenance and pretreatment cost: 5% investment cost [69]. Operating workers and supervision: 1 manager, 1 electrician, 2 mechanical fitters, 2 laborers, 1 laboratory technician [66].
Spare Parts:	-----	-----	Maintenance and spare parts: 3% of direct capital (investment) less membranes [58].
Chemicals:			7% of total production cost [68]
Acid (ppm)	340 [51]	150(j); [71]	15-40 [71]; 30 [72], 103 [70]
Antiscalant (ppm)	5 [71]; 2.25 [72]	11(f; i) [51]; 12.5(g; i) [51]; 5(i); [71]; 15(j); [71]	15 [51]; 5 [71]
Coagulant (ppm)	-----	-----	7.5 [51]
Chlorine (ppm)	-----	-----	3.5(h); [51]; 2 [72]
Sodium bisulfite (ppm)	-----	-----	9 [72]
Antifoam (ppm)	1.6 [51]	Used [71]	-----
Membranes replacement rate (%/year):	No used	No used	33 [51, 70], 20 [59,69], 10 [58]

(a): 272.7 kPa gauge reduced at 31 kPa absolute, 70ºC
(b): Without energy recovery
(c): With energy recovery
(d): Spiral wound RO
(e): Hollow fiber RO
(f): Polyphosphate scale inhibiting additive,ID104 [51]
(g): Scale cleaning chemical ID-108, sulphamic acid [51]
(h): Sodium hypochlorite
(i): Low temperature
(j): High temperature

Table II. A summary of the data in relation to the three methods

COSTS	DESALINATION PROCESS					
	MSF		MED		RO	
CAPITAL COSTS, in 10^6 U$S						
	CAPACITY OF PLANT		*CAPACITY OF PLANT*		*CAPACITY OF PLANT*	
	≈32,000 m^3/d	**≈6,000 m^3/d**	**≈32,000 m^3/d**	**≈6,000 m^3/d**	**≈32,000 m^3/d**	**≈6,000 m^3/d**
Plants installed:	34.5 [54]	-----	32.4 [54]	-----	28.7(a); [54]; 25.5(b); [54]	-----
Seawater intake and outfall:	2.8 [54]	-----	2.6 [54]	-----	2.0(a); [54]; 1.8(b); [54]	-----
Foundations and buildings, 15%:	5.6 [54]	-----	5.2 [54]	-----	4.6(a); [54]; 4.1(b); [[54]	-----
Financing during construction,	4.3 [54]	-----	4.0 [54]	-----	4.6(a); [54]; 4.1(b); [54]	-----
Engineering and contingency,	4.3 [54]	-----	4.0 [54]	-----	4.6(a); [54]; 4.1(b); [54]	-----
Total:	51.4 [54]	9.00 [51]	48.3 [54]	14.0 [51]	42.4(a) [54]; 37.7(b) [54]	7.0 [51]
PRODUCT WATER COST, in U$S/$m^3$						
Variable costs						
Energy:	0.54 [51]		0.17 [51]		0.44 [51]	
Heat:	0.242 [54]		0.219 [54]		0.000 [54, 59]	
Power:	0.109 [54]		0.07 [54]		0.128(a) [54];0.106(b) [54]; 0.316(c) [59]; 0.39(d) [59]; 0.35 [70]; 70% of total production cost [68]	
Operation & maintenance:	0.126 [54]. Maintenance 2%/year of capital costs = 0.08 [51]		0.126 [54]. Maintenance 2%/year of capital costs = 0.13 [51].		0.126 [54]; 0.057(c; d) [59] Maintenance: 2%/year of capital costs (= 0.06) [51]; 0.15 [70]; 23% of total production cost (with membranes and other parts included) [68]. Maintenance and pretreatment: 5% of capital cost [69].	
Spare Parts:	0.082 [54]		0.082 [54]		0.033 [54]; 0.03(c; d) [59] Maintenance and spare parts: 3% of capital cost [58].	
Chemicals:	0.024 [54]; 0.06 [51]		0.024 [54]; 0.03 [51]		0.047 [54]; 0.037(c; d) [59]; 0.03 [51, 69]	
Fixed costs						
Membranes replacement:	0		0		0.110(a) [54]; 0.098(b) [54]; 0.120(c; d) [59]; 0.15 [51]; 0.14 [69]	
Capital charges:	0.461 [54]; 0.58 [51]		0.433 [54]; 0.89 [51]		0.380(a) [54]; 0.338(b) [54]; 0.093(c; d) [59]; 0.45 [51]	
Total:	1.043 [54]; 1.26 [51]		0.953 [54]; 1.23 [51]		0.823(a) [54]; 0.747(b) [54]; 0.653(c) [59]; 0.727(d) [59]; 1.13 [51]; 0.68-0.81 [67]	
Typical operating costs for 11,000m^3/d R O units [68]:	-----		-----		0.4-0.6 [68] (15% Labor; 7% Chemicals; 70% Energy; 8% Maintenance membranes and other parts) [68].	

(a): Without energy recovery (b): With energy recovery (c): Spiral wound RO (d): Hollow fiber RO

In the lower power module, the reactor coolant circulates by natural convection. The natural circulation flow takes place by the difference between the density of the coolant in the upward-hot branch and the one in the downward-cold branch. This difference provides the coolant driving force to circulate.

The steam generators are the "once-through" helical tubes type. In these generators the flows of the primary and secondary systems circulate in countercurrent flow. The secondary fluid circulates upwards within the tubes. It flows into the tubes as liquid-water and it reaches the exit as overheated vapor.

Self-pressurizing of the RPV -in the steam dome- is produced by the vapor-liquid equilibrium. The outgoing temperature of the core is the saturation temperature of the primary system pressure. Then, the pressure interference is highly damped.

The design of the security systems fulfills the requirements of the regulations of the nuclear industry as for redundancy, independence, physical separation, diversification and failure into a safe state.

CAREM safety systems must guarantee no need of active actions to mitigate accidents for a long period.

CAREM has two different and independent shutdown systems. These systems are designed to shutdown and to maintain the reactor core sub-critical. They are activated by the protection system reactor. The first system is designed to shutdown the core reactor by dropping neutron-absorbing elements into the core by the action of gravity. The second shutdown system is based on the injection of borated water to the core, also by the action of gravity.

The residual heat of the core, in station blackout, is removed by passive principles (natural convection) through the residual heat removal system. This system transfers this energy to the pressure suppression pool.

CAREM has an emergency injection system to prevent core exposure in case of loss of coolant accident (LOCA). This system assures the correct refrigeration of core reactor without electric power supply.

The RPV integrity is additionally ensured by three safety relief valves. They protect the RPV against overpressures and each valve has 100% of the necessary relief capacity.

CAREM has a containment isolation -pressure-suppression type- to retain the eventual liberation of radio-active materials. Its design is such that after having begun any unlike accident with loss of coolant, and without any external action, the pressure inside stays below the design pressure.

In the Figure 3 an outline of the containment and safety systems for module of lower power is are shown.

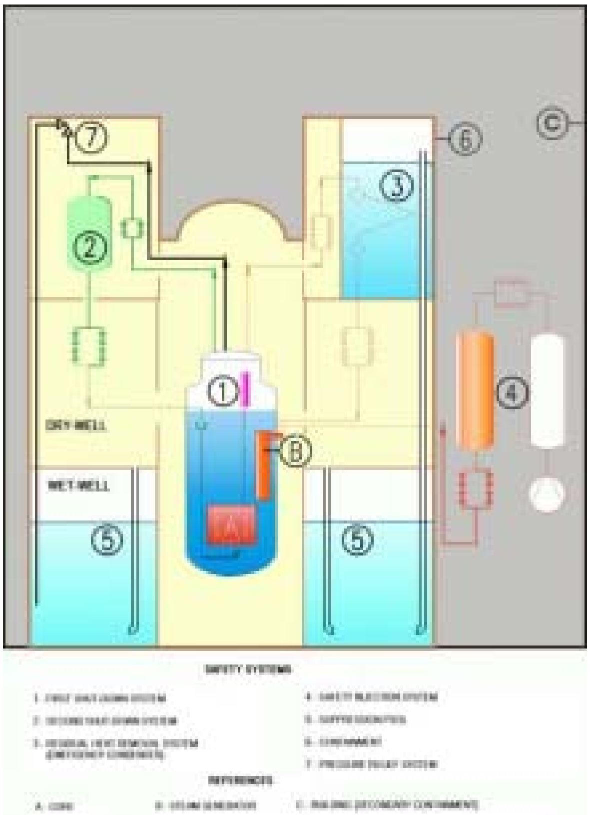

FIG. 3. Containment and safety systems.

5. SITE WITH SCARCITY WATER AND THE FEATURES OF A POSSIBLE SOLUTION

5.1. Selection of region and site

5.1.1. Argentinian site

Already obtained data' analysis in section 2 indicates that installing a nuclear desalination plant in Puerto Deseado –Argentinian region- would be a good solution due to the scarcity of fresh water in the place. The main reasons for this selection are:

— A severe scarcity of fresh water.
— Many possibilities of industrial growth.
— The importance of to install a power plant in the city to increase the reliability of its energy supply.

The relevant parameters of Puerto Deseado have been collected. The most important ones are:

5.1.1.1. Site's communication and accessibility [43]

The city is connected to National Highway N°3 by National Highway N°281, along 125 km. The most important highway in the Argentinian South is the N°3. This highway goes from Buenos Aires to Ushuaia. A local airport communicates the city with Río Gallegos and Comodoro Rivadavia. There are also many long distance public transportation companies.

5.1.1.2. Geology and Volcanism [43]

In Puerto Deseado three different rocky complexes can be distinguished: a solid quartzitic porphyry associated with tufas; on the former solid eroded surface there are deposits of an old marine transgressions and on the last terrace, deposits of Patagonian pebbles were settled on the upper Tertiary and the Quaternary. There are no volcanic manifestations.

5.1.1.3. Seismology and Earthquakes [74]

Puerto Deseado is far from any earthquake areas. The seismic risks of Puerto Deseado, according to the seismic zoning of Argentina, is very reduced (the lowest qualification in the scale). The maximum acceleration of the ground is 0.04, expressed as a fraction of the acceleration of gravity.

5.1.1.4. Extreme meteorological events [75 – 77]

Among the extreme meteorological phenomena are hurricanes, tornadoes, severe storms, tropical storms and typhoons. Only severe storms and tornadoes take place in Argentina. But they usually happen in the north of the parallel 45º south. Puerto Deseado is in the south of that parallel, then tornadoes and strong storm probabilities are very low.

5.1.1.5. Meteorological information [78]

The climate of Puerto Deseado oscillates between temperate and moderate cold.

— Temperatures: The maximum temperature took place in January and it was of 34.6°C, and the minimum temperature took place in June and it was of -10°C. The seasonal average temperatures are:
  - Summer:15.5°C
  - Autumn: 7.8°C
  - Winter: 4.9°C
  - Spring: 13.5°C
— Humidity and wet-bulb temperature: Average humidity relative is 66% and the seasonal average wet-bulb temperatures are:
  - Summer: 11.3ºC
  - Autumn: 5.1ºC
  - Winter: 3.6ºC
  - Spring: 9.5ºC
— Rainfalls and snowfalls: Rainfalls reach low annual values: 230 mm (ten years average). Rains frequency is higher in winter. The most frequent snowfalls are in June and July, but it does not snow every year; furthermore when it snows, it melts quickly.
— Winds: The prevalent winds are from Western quadrant, with an average speed of 33 km/h and their frequency is higher in spring.
— Pressure: Average pressure at sea level: 1,008.7 hPa

5.1.1.6. Oceanographic data [79]

— Tide regimen: Semidiurnal.
— Maximum width: 5.88 m.
— Medium width: 3.71 m.

5.1.1.7. Seawater quality [80]

— Seawater temperatures: The maximum temperature took place in January and it was of 16.6°C, and the minimum temperature took place in July and it was of 2°C. The seasonal average temperature are:
  - Summer:13.2°C
  - Autumn: 8.6°C
  - Winter: 5.3°C
  - Spring: 10.2°C
— Average seawater salinity (total dissolved solids, TDS): 33,300 ppm

5.1.1.8. Cooling water

An adequate supply of cooling water can be available.

5.1.2. Selected region in the rest of Latin America

According to the analyzed information it is considered that the most convenient region to carry out a study of feasibility of installation of a seawater-desalination plant using nuclear energy is the corresponding to the Dry-Pacific hydrographic system, coastal area.

The reasons that would favor this selection are, among others, that Chile as well as Peru, both countries part of that region, has developed an important activity in the nuclear area for many years; they also have nuclear research reactors. On the other hand, Chile has important coal reserves, but it does not have petroleum and gas big reserves. Besides, the Caribbean islands habitually have natural extreme events, such as tornadoes and hurricanes, what would increase the costs for a first comparative evaluation.

5.2. Selection of desalination technology

For Puerto Deseado, Argentinian site, technology selection was carried out according to:

— The fact that Puerto Deseado is a small city with many possibilities of industrial growth and therefore with a growing demand of fresh water. Other characteristics of the place, such as the quality of seawater, the climatic conditions, the required quality of the water and the available infrastructure were also evaluated.
— World tendencies and technological advances of desalination processes, and available infrastructure.
— Argentinian experience in operation of desalination plants.

According to that, RO is considered the most convenient technology for the selected site since:

— RO allows using modular units and then the plants are more flexible. This characteristic is important for the selected place. Puerto Deseado is a small city and it will be possible to begin the operation of the installation with a module and to continue adding others according to what its industrial development requires.
— RO has high recovery rate, high availability, low energy consumption, simple operation and it only requires a little supervision.
— Advances in materials' technology, unit operations and process control for RO processes have improved plant efficiencies and its investments to obtain lower costs in recent times.
— RO has a very flexible production's ratio of power to water. Their interconnection requirements with the energy source favor the selection, because the installation to be evaluated includes the production of drinkable water and energy.
— Argentina has experience in operation of small units of RO.

5.3. Selection of nuclear power plant CAREM as energy source

In the analysis of the possible electric sources, the following has been kept in mind:

— Puerto Deseado has shortage of fresh water and the installation of an electrical power plant would also increase the reliability of the electric supply of the site. Therefore, the cogeneration of water and electricity would be a good solution for both problems.
— The technology selected for seawater desalination in Puerto Deseado is RO and therefore the coupling between the electric power plant and the desalination plant is simple and it only requires an electrical connection. This implies that:
  - No modification is required in the design of the secondary refrigeration system of the electric power plant.
  - Independent operation of the electric power plant and of the desalination plant gives flexibility to the system, allowing to increase if necessary, the size of the desalination plant without any modifications in the electric power plant. This is a good point for a city in growth.

— Puerto Deseado is today a city without environmental contamination and with many possibilities of tourist expansion in the whole adjacent region. The negative effects on the environment of burning fossil fuels to generate electric power are very well-known, as well as the advantages of nuclear power plants in this aspect because they have no emission of greenhouse gases and no environmental pollution.
— Installing a nuclear power plant in Puerto Deseado will contribute to the diversification of the energy sources.
— A project of nuclear desalination in Puerto Deseado would have the additional advantage of a spin-off in industrial development in the city.
— Argentina has an important experience in the nuclear area with highly satisfactory results in the electric generation of nuclear origin.

According to these reasons a nuclear power plant as an energy source for a desalination plant in Puerto Deseado is a good option.

On the other hand Argentina has a project of an innovative reactor CAREM. This nuclear power plant has the inherent advantages of an innovative nuclear power plant, such as:

— CAREM nuclear power plant has a design that contributes to a high level of safety such as integrated primary cooling system; primary cooling by natural circulation, self-

pressurized and passive safety systems. CAREM safety system guarantees no need of active actions to mitigate the consequences of human errors or accidents during a long period.
— CAREM nuclear power plant is a small plant and has the advantages such as a big potential of design standardization, series production and shop fabrication of equipment. Therefore, it is possible to obtain a low construction cost. It is very well known that the competitiveness of nuclear desalination option increases significantly if the capital cost decreases.
— Higher load factors, will allow increase the reliability of the electric supply of the site.
— CAREM nuclear power plant as it has been developed up to now, including all secondary circuit, can be directly taken as a fundamental part of the solution here it has proposed, without signification modification or adaptation for this purpose. However it remains to be analyzed if the present nominal reactor power is the best or if some increase could be desirable.

Therefore, the CAREM nuclear power plant was also considered an attractive, technically feasible and safe alternative for electricity and potable water production according to Puerto Deseado site characteristics and its inherent advantages of innovative nuclear power plant.

6. SUMMARY

Nuclear desalination is a possible solution to Latin American region's ongoing water scarcity.

According to IAEA's recommended method [52], in this paper four regions of Latin America, for which the seawater desalination would be a solution to its shortage of fresh water, were selected. One of them is in Argentina and the others are in rest of Latin America.

The selected regions, in Argentina and in the rest of Latin America, have shortage of fresh water and energy. These regions have seawater and important potential of their population's growth and industrial activity. In short, these regions are interesting places to carry out an integral evaluation to install a nuclear desalination plant.

In Argentina the selected site is Puerto Deseado and in the rest of Latin America the selected region is the Dry-Pacific hydrographic system.

On the other hand, now seawater desalination is a promissory solution to obtain driking water. Thus, in recent years RO technology has become a more attractive technology for seawater desalination because of its lower membrane costs, its lower energy requirements and the efforts to obtain high reliability and availability for the process.

Technology selection for Argentina was carried out according to: particular characteristics of Puerto Deseado, world tendencies and technological advances of the desalination processes, and some economic data. According to that, RO is the technology that is considered as the most convenient for the selected site.

Regarding the energy supply, Puerto Deseado is connected to the Patagonic Interconnected System with an electric line of 132 kW from Pico Truncado to Puerto Deseado. This city is situated in the end of this line. In order to increase the reliability of supply, it is very important that the city can have a power plant.

The CAREM nuclear power plant is considered an attractive, technically feasible and safe alternative for electricity and potable water production according to Puerto Deseado site characteristics and its inherent advantages of innovative nuclear power plant, such as the higher load factors and the environmental considerations.

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OPPORTUNITIES FOR USING THE INNOVATIVE NUCLEAR REACTORS IN ARMENIAN ENERGY SECTOR LONG TERM PROGRAMME DEVELOPMENT

A. GEVORGYAN

Ministry of Energy of Armenia
Yerevan, Armenia

Abstract. The purpose of the present paper is to demonstrate the importance of nuclear energy development in Armenia when considering the long-term energy planning, taking into account the specific conditions and tendencies, which are formed and developed in economy of Armenia and, in particular, in fuel-energy complex of the country. When developing the long-term program, the main factors among others considered were assumed to be the energy independence and energy security of a country, and not only the least –cost factor, as it was usually done before. When that program was under development, such social aspects as application of the infrastructure existing within the relevant sphere, and financing of decommissioning of existing units of the AMP were also took into consideration. The studies performed have shown that implementation of innovative medium size reactors would enable the energy sector of Armenia to meet all those requirements. The issues of environmental protection were also taken into consideration when developing that program.

1. ENERGY SECURITY AND INDEPENDENCE

As it was mentioned above, the principles of energy independence and energy security were laid in the base of the long-term planning of Armenian energy development, and it is obvious that in the countries with complete absence of domestic fuel resources (deposits), such as Armenia, the issues of provision of energy security and independence take on primary priority.

The concept of energy security for Armenia under the existing conditions can be formulated in the following way - the country should have guaranteed ability for the reliable energy supply for all requirements of society both under the sustainable development and in extreme conditions. Ensuring the energy security is the main task and responsibility of all the state institutions without exceptions. This should include the participation of private and public organizations of Armenia, too.

In order to ensure the country’s energy security, it is necessary to guarantee its energy independence. It means that the country’s energy sector should be minimum dependent on the imported fuels, that is, it should be achieved the maximum utilization of the domestic energy sources.

Taking the above-mentioned principles as a basis, we have modeled the Armenian electric-energy sector long-term development taking into account the future needs to cover the electricity demand forecasted. Two options were considered: the energy sector development including the nuclear energy scenario, and the option without the nuclear way of electricity generation, called combined cycle scenario.

Summarizing the experience of energy crisis, lasted in Armenia during 1992-1996, we can assert that, upon having 40% energy independence, the normal functioning of practically all the life-support systems of Armenia in wide range of emergency situations can be assured. And only restarting Unit 2 of the ANPP made it possible to stop that crisis evolution and enable the country to move toward the further economic development.

Figure 1 shows the level of energy independence of Armenian energy sector on the implementation of "nuclear" scenario. It can be seen that sufficiently high level of energy independence and, consequently, energy security, may be obtained upon nuclear energy development.

For the combined cycle scenario, the decommissioning of Unit 2 of Armenian NPP will badly affect the level of country's energy security and independence (Figure 2).

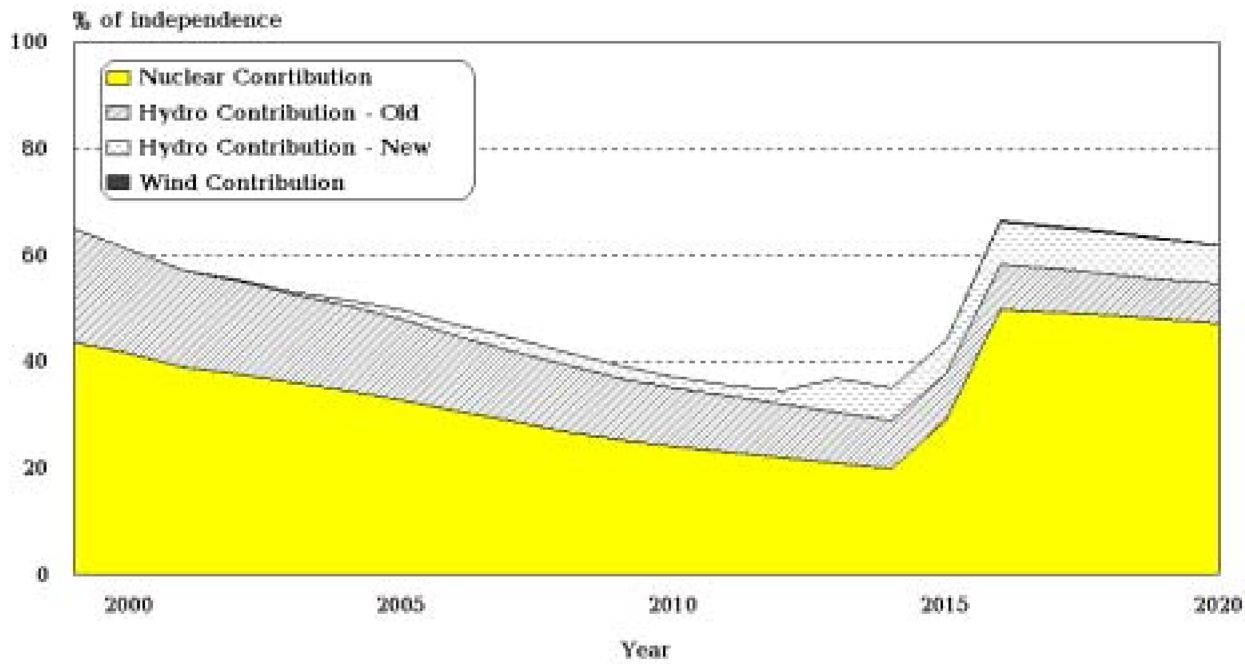


FIG. 1. Percentage of Armenian electricity sector independence from imported energy sources - Reference demand-Nuclear scenario.

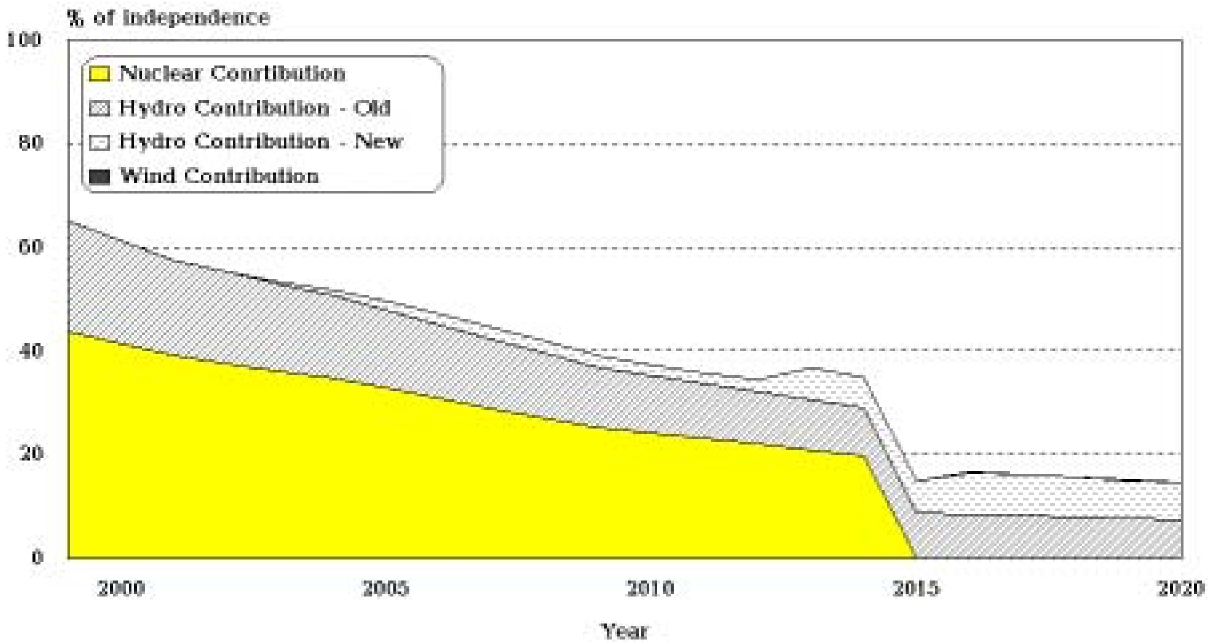


FIG. 2. Percentage of Armenian electricity sector independence from imported energy sources - Reference demand - CC scenario

2. ECOLOGICAL ASPECT

Armenia is faced with a number of environmental problems, but has severely limited resources to address them. The Government has taken a number of steps to ease environmental degradation working closely with many international environmental protection organizations and other institutions to strengthen its capacity to improve the ecological situation in Armenia.

Among the various factors responsible for degradation of natural environment, the most polluting one is related with the use of energy. Environmental aspects are the most important components of the energy security.

The environmental implications of the energy sector of Armenia has been examined because, although some about 30 - 35% of the present electricity generation is based on fossil fuels (natural gas), it results in emission of 1 267 213 tons of CO_2, 3355 tons of NOx, 960 tons of CO; and 9 tons of SO_2. These emissions and other wastes are expected to increase in the coming decades due to the foreseen increase in electricity production, which will be based on fossil fuels.

As it is widely known, nuclear power plants do not produce gases such as CO_2 CO, SO_2 and NOx, which are responsible for acid rain and global warming. Although, some radioactive materials are released to the environment during normal operation of nuclear power plants and other nuclear fuel cycle facilities, the amounts released are very small and strictly kept within the permissible limits. So, we can say that the option with combined cycle scenario, from this point of view, is not as preferable as that with the nuclear scenario.

Moreover, continuing to develop option with the nuclear scenario would allow preserving the Armenian national value – lake Sevan. The problem of Sevan, the only large water reservoir (basin) in the territory of Armenia, takes special place among the complex of ecological issues of Armenia. The level of the lake was crucially decreasing because of the use of Sevan hydro potential for energy purposes during the energy crisis described above in spite of the Governmental program on Sevan lake protection. Only after the restart of Unit 2 of Armenian NPP, the water outflows from Sevan Lake were restricted to the volume that was necessary for irrigation needs.

Therefore, the environmental aspect of energy sector was also taken into consideration when planning its future development. The ecological factors were included into the study when modeling scenarios of long-term energy development.

3. SOCIAL ASPECT

At present, the ANPP is not only the energy generating facility, the power plant with all the adjacent supporting organizations has at its disposal a great potential of several thousands of high-skill specialists, the whole generation brought up during the long period of establishment of nuclear energy in Armenia.

The decommissioning of the ANPP will mean not only the closing of this energy object, but also the shutdown of a whole science-intensive and high technology direction of this branch of country's economy. So, the most part of those qualified specialists will have to change their work positions due to the abrupt reduction of employees required for the final stage of the ANPP operation.

Though, in case of realization of an option of energy sector development with the nuclear scenario, it is expected that more than 10.000 workers will be employed in the construction process. Considerable part of industry and transport infrastructure of the country will be involved. For the country that suffers a transition period, such huge construction may be a locomotive for the whole economy.

4. SENSITIVITY ANALYSIS OF IMPLEMENTING THE NUCLEAR SCENARIO

The sensitivity analysis was performed for both scenarios of development of Armenian energy sector. It is obvious that, in case of realization of nuclear scenario, the key project of the program will be the construction of new units of Armenian NPP. At that, the problem of commissioning period for new units will be tightly connected with the decommissioning period of existing units of the ANPP. In this connection, the conducting of sensitivity analysis of the impact of commissioning periods of NPP' new units with simultaneous decommissioning of exhausted units is of special inters, and the present value (PV) is accepted as a main criteria of sensitivity analysis.

On the assumption of the calculation results for different scenarios, the PV values differ slightly in rather wide range of commissioning periods for new capacities. However, the "late" commissioning (2018 or later) of even one of the planned new nuclear units is improbable, since it may occur only in case of possibility of prolongation of operation resource of Unit 2 of the ANPP for several years, or on the condition of timely creation of other capacities, alternative to Unit 2.

On the other hand, the following circumstances represent the obstacles for "early" commissioning the new units:

— Project capital intensity (financing provision),
— Preparation and decommissioning of existing units of ANPP,
— Development of intra-system and inter-system transportation networks,

Thus, the best timing on the commissioning of new nuclear unit must be accurate coordinated with the shutdown of operating Unit 2 of the ANPP. In other words, the optimal energy development plan may be achieved upon the acceptance of the postulate on consideration of construction of new nuclear power unit as alternative to the existing ones.

The implementation of nuclear option would also make it possible to include the costs of decommissioning of the old units of the ANPP into the tariff on the electricity generated by the new nuclear unit.

5. CONCLUSION

The analysis performed has shown that the sustainable energy long-term development in Armenia can be achieved provided that the nuclear scenario will be implemented as the preferable, in view of all aspects, of the two above-mentioned scenarios considered. This may be ensured in case of utilization of innovative nuclear reactors with high-level operational safety and economic indicators.

FIXED BED SUSPENDED CORE NUCLEAR REACTOR CONCEPT

F. SEFIDVASH

Nuclear Engineering Department
Federal University of Rio Grande do Sul
Porto Alegre, Brazil

Abstract. The fixed bed nuclear reactor (FBNR) is essentially a pressurized light water reactor (PWR) having spherical fuel elements constituting a suspended reactor core at its lowest bed porosity. The core is movable thus under any adverse condition, the fuel elements can leave the reactor core naturally through the force of gravity and fall into the passively cooled fuel chamber or leave the reactor all together entering the spent fuel pool. It is a small and modular reactor being simple in design. Its spent fuel is in such a convenient form and size that may be utilized directly as the source for irradiation and applications in agriculture and industry. This feature results in a positive impact on waste management and environmental protection. The principle features of the proposed reactor are that the concept is polyvalent, simple in design, may operate either as fixed or fluidized bed, have the core suspended contributing to inherent safety, passive cooling features of the reactor. The reactor is modular and has integrated primary system utilizing either water, supercritical steam or helium gas as its coolant. Some of the advantages of the proposed reactor are being modular, low environmental impact, exclusion of severe accidents, short construction period, flexible adaptation to demand, excellent load following characteristics, and competitive economics.

1. INTRODUCTION

The solution to the ever increasing demand for energy to satisfy the needs of growing world population and improving its standard of living lies in the combined utilization of all forms of energy. Nuclear energy produced safely and economically has an important role in solving the world energy problem. The public objections to nuclear energy most often expressed are reactor safety, cost and nuclear waste disposal. The proposed reactor concept is to meet the requirements of a modern reactor.

At the end of 2000, there were 438 nuclear power reactors in operation in 31 countries around the world, generating electricity for nearly 1 billion people. They account for approximately 17 percent of worldwide installed base capacity for electricity generation.

About 30% of the world’s primary energy consumption is used for electricity generation, about 15% is used for transportation, and the remaining 55% is converted into hot water, steam and heat. Non-electric applications include desalination, hot water for district heating, and heat energy for petroleum refining, for the petrochemical industry, and for the conversion of hard coal or lignite. For non-electric applications, the specific temperature requirements vary greatly. Hot water for district heating and heat for seawater desalination require temperatures in the 80° to 200° C range, whereas temperatures in the 250° to 550° C range are required for petroleum refining processes and about 800° C are necessary for coal gasification processes.

The problem of energy resource availability, climate change, air quality, and energy security suggest an important role for nuclear power in future energy supplies. Nuclear power plant technology has evolved as three distinct design generations: (1) Prototypes, (2) Current operating plants, and (3) Advanced reactors. The next generation of nuclear energy systems must be licensed, constructed, and operated in a manner that will provide a competitively priced supply of energy, keeping in consideration an optimum use of natural resources, while addressing nuclear safety, waste, and proliferation resistance, and the public perception concerns of the countries in which they are deployed. Recognizing both the positive attributes and shortcomings of the prior generations of reactor designs, it is now time to lay the groundwork for a fourth generation nuclear reactors.

The U.S. Department of Energy's Office of Nuclear Energy, Science and Technology has engaged governments, industry, and the research community worldwide in a wide-ranging discussion on the development of next-generation nuclear energy systems known as "Generation IV". It refers to the development and demonstration of one or more Generation IV nuclear energy systems that offer advantages in the areas of economics, safety and reliability, sustainability, and could be deployed commercially by 2030.

2. REACTOR DESCRIPTION

The fixed bed nuclear reactor concept is modular in design such that any size of reactor can be constructed from the basic module (Figure 1). It is an integrated primary system design. The basic module has in its upper part the reactor core and a steam generator and in its lower part the fuel chamber. The core consists of a 25-cm diameter tube in which, during reactor operation, the spherical fuel elements are held together in a fixed bed configuration forming a suspended core. The fuel chamber is a 10-cm diameter tube, which is directly connected underneath the core tube. A steam generator of the shell and tube type is integrated into the upper part of the module. A neutron absorber shell slides inside the core tube, acting similarly to a control rod.

The pump circulates the coolant inside the module moving up through the fuel chamber, the core, and the steam generator and thereafter flows back down to the pump through the concentric annular passage. At the maximum or terminal fluidizing velocity, the coolant carries up the fuel elements from fuel chamber into the core. In the shut down condition, the suspended core breaks up and the fuel elements leave the core and fall back into the fuel chamber by the force of gravity.

The 8-mm diameter spherical fuel elements are made of slightly enriched uranium dioxide, cladded by zircaloy for normal design, and stainless steel when using supercritical steam. The fresh fuel elements are fed into the reactor core from the top of the module. The spent fuel leaves the module through a valve provided at the bottom of the fuel chamber. The valve is operated by a hydraulic system allowing the spent fuel to be discharged from the fuel chamber into a permanently cooled storage tank. The reactor is provided with a pressurizer system to keep a constant pressure, and a depressurizer valve which leads the steam to the condenser for reducing pressure to allow opening of the valves for refueling.

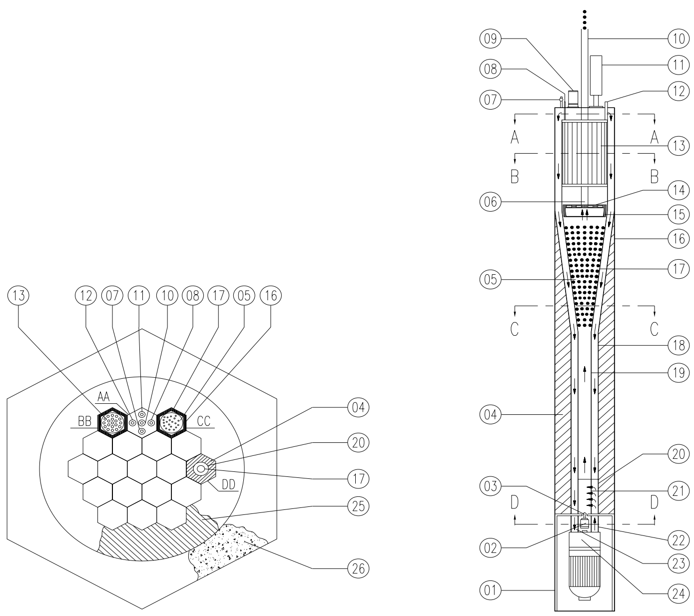
Any hypothetical accident will cut-off power from the pump causing the fuel elements to leave the core and fall back into the fuel chamber by the force of gravity where they remain in a highly sub critical and passively cooled condition. The fuel chambers are cooled by natural convection transferring heat to the surrounding air or water pool.

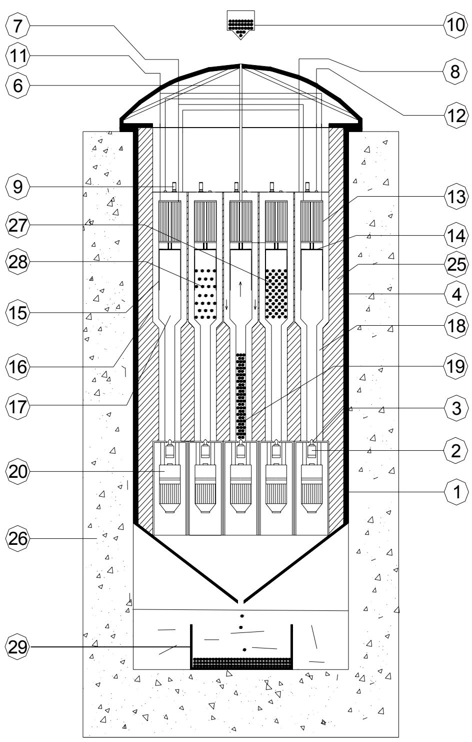
A detailed heat transfer analysis of the fuel elements has shown that due to a high convective heat transfer coefficient and a large heat transfer surface to volume ratio, the maximum power extracted from the reactor core is not limited to the material temperature limits, but to the maximum mass flow of the coolant corresponding to the allowed porosity.

The core tube is designed in a slightly conical shape in such a way the coolant velocity decreases along the height of the reactor. The slow and continuous reduction of velocity has a compacting effect securing a stable fixed bed.

The concept is very flexible and have the possibility of devising many types of design. The proposal for the first stage is to have a “Fixed” Bed Nuclear Reactor, where the fuel elements fluidize out of the fuel chamber into the core becoming a fixed bed being at its lowest porosity, forming a static “suspended core” in the upper part of the module. Therefore, in this case there would not exist any collisions between the fuel elements and eliminates the concerns about the fuel performance. The fully fluidized bed design would be implemented later on when the concerns about the fuel reliability in the fluidized condition will be confirmed.

Each module has independently a pump but a single refueling machine serves all the modules. A crown type header on the top of the reactor joins and connects all the modules to a common system where the entering and outgoing fluids are integrated into a unique system.





(1) structural support;
(2) hydraulic valve opener;
(3) fuel discharge valve;
(4) graphite jacket;
(5) reactor core;
(6) level limiter shaft;
(7) depressurizer;
(8) steam exit;
(9) level limiter drive;
(10) fuel feed;
(11) pressurizer;
(12) water entrance;
(13) steam generator;
(14) level limiter;
(15) absorber shell;
(16) hexagonal channel;
(17) fluidization tube;
(18) circular channel;
(19) fuel chamber;
(20) distributor
(21) entrance perfurations;
(22) coolant entrance;
(23) coolant exit;
(24) primary pump;
(25) reflector;
(26) biological shield,
(27).fixed bed core,
(28) fluidized bed core,
(29) spent fuel pool.

FIG. 1. Schematic design of the reactor.

3. DESIGN OPTIONS

This reactor concept gives the possibility of various design options to be introduced.

3.1. Fixed bed with liquid water as coolant

The reactor in its simplest form consists of a pack of spherical fuel elements forming a fixed bed. The fixed bed is suspended and is kept cooled by the water which flows up in the core. Therefore, in this case there would not exist any collision between the fuel elements which eliminates the concerns about the fuel performance.

3.2. Fixed bed with supercritical steam as coolant

The concept of a direct cycle reactor operating at supercritical pressure is attractive for improving the thermal efficiency drastically to enhance the resulting environmental protection. The reactor combines the fixed bed concept with the idea of using direct cycle reactor operating at supercritical pressure proposed by Oka [4]. The supercritical steam is used as the reactor coolant. The critical pressure of water is 221 bars. When the reactor operates at 250 bars, the supercritical water does not exhibit a change in phase and the concept of boiling does not exist. The water density decreases continuously with temperature.

The coolant entering temperature, on the lower part of the bed, is 310 ºC and the exit temperature, on the upper part of the bed, is 416 ºC. Therefore, the water density decreases continuously from 0.725 to 0.137 g/cm^3 along the bed. This is an important factor in causing the bed to become a more stable and having fixed core. The recommended pressure of 250 bars is due to the smooth and mild variation of density with pressure in this region resulting in stability of flow in the core. The power production is much higher in this option as the difference in inlet and outlet enthalpy is much higher than a simple pressured or even boiling reactor. The plant thermal efficiency is estimated to exceed 40% which is about 20% higher than the conventional pressurized water reactors. The turbines will be smaller compared with the light water reactors by adopting the supercritical steam as the coolant. The superheated steam is fed directly into the turbine. The steam-water separation is not needed for direct cycle reactor. Some other advantages of such a choice besides the high thermal efficiency, will have smaller turbine, no steam generators, and reduced waste heat.

3.3. Fixed bed with helium gas as coolant

In this option, the fixed bed is cooled by helium gas yielding all the advantages of a gas cooled reactor including high efficiency and utilization of direct gas turbine. In this case we have a fast nuclear reactor system.

3.4. Fluidized bed with water as coolant

The power density may significantly be increased by fluidizing the bed. The increase in turbulence of the coolant will allow a significant increase in power generation. Such a study was related elsewhere [1-3]. In this case the effects of flow on the homogeneity of the fluidized bed porosity and physical interaction between fuel elements need to be further studied.

3.5. Fluidized bed with supercritical steam as coolant

This option still takes advantage of further increase in heat transfer rate and higher temperature steam production leading to still a more efficient system.

4. WATER DESALINATION

The combination of a nuclear power reactor with water desalination plant can most economically be utilized because the higher the temperature and pressure of steam used in a turbine, the lower the cost of electricity produced. For fractional distillation, on the other hand, steam at low temperature and pressure is needed and the greater part of input heat is the latent heat of steam. Therefore, the power production and desalination systems may be advantageously combined. For example, to produce 300 MWe of electricity and 570,000 m^3/day of water separately, one needs about 2600 MWt of energy, but to produce the same amount of electricity and water in a dual purpose plant one needs only about 1850 MWt of energy, a saving of about 40% in energy consumption. A 100 MWe fixed bed suspended core nuclear power reactor when employed in a dual purpose plant for producing both power and water will produce 70 MWe of electricity and about 130,000 m^3/day of desalted water.

5. WORLD NUCLEAR ENERGY COMPANY (WONEC)

It is desirable that a World Nuclear Energy Company (WONEC) be formed to become a catalyst in organizing and coordinating the world-wide existing and to-be-created scientific, technological and industrial elements in order to supply the world with clean and safe nuclear energy. WONEC can supply the world with the proposed inherently safe nuclear reactors and will be responsible for its entire fuel cycle. It is to function as a commercial as well as scientific venture with highest international standard. Its shares can be freely traded in the international financial market. WONEC is to operate under auspices of the International Atomic Energy Agency. WONEC by the nature of its policies, compositions, and adopted legal and ethical values will have the conditions and credibility to supply nuclear energy and create public confidence in nuclear energy.

The research and development needed are to be performed by the participation of all the interested companies and research organizations of the world. It is to be done in a true spirit of international cooperation and service to humanity. Nuclear technology no longer is a monopoly of any single or group of nations and to various degrees most nations posses them. A new method of consultation and decision making process will be applied in order to safeguard the interest of all. The participants will not need to control WONEC to have their interests guaranteed. The project is to remain a totally scientific, industrial, and economic venture, avoiding dominating national politics, and remain in conformity with the spirit of the new age and the presently growing international desire for world peace.

The developing countries are expected to show great interest in participating in WONEC since in this way they will acquire nuclear power without the fear of being exploited by the vendors or making very large investments for an independent national nuclear program. The industrialized countries are expected to support the idea as well, since their existing reactors are not selling well, and by participation in WONEC they will benefit from the sales of their technologies to WONEC and partake in a very large nuclear reactor sales market worldwide. Also the problem of nuclear proliferation which is of their great concern will be under control. The WONEC is to operate in the spirit of the new era providing the citizens of the world with clean and safe nuclear energy.

6. POTENTIAL OF THE CONCEPT FOR MEETING THE GENERATION IV GOALS

The simplicity of this reactor design results in short lead time, low capital and power generation costs. The likelihood of core damage is extremely low and moreover the suspended core can be removed from the reactor system at any desired instant. The inherent safety characteristic of the reactor makes it tolerant against any human errors or equipment failure. It may even be considered as safe against terrorist activities.

As the spent fuel elements are appropriate to be used directly as radiation source in industrial and agricultural applications. They may not be considered as nuclear waste thus increases the public acceptance of waste solution. If thorium were used in this reactor concept, it will make it less attractive to nuclear proliferates. The smallness of the reactor brings along with it numerous advantages widely described in the literatures.

A detailed analysis show that to a great extend the proposed nuclear reactor meets the requirements of the fourth generation nuclear reactors, namely: (1) Provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production. (2) Minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment. (3) Excel in safety and reliability. (4) Have a very low likelihood and degree of reactor core damage. (5) Eliminate the need for offsite emergency response. (6) Have a clear life-cycle cost advantage over other energy sources. (7) Have a level of financial risk comparable to other energy projects.

7. CONCLUSIONS

The FBNR uses PWR technology in its simplest form. For its deployment there is no new technology to be developed. At a very low cost, its feasibility can be fully demonstrated, and in less than 5 years the reactor can enter the market. It is a small and modular reactor being simple in design. Its spent fuel is in such a convenient form and size that may be utilized directly as the source for irradiation and applications in agriculture and industry. This results in a great positive impact on waste management and environmental protection.

The advantages of the proposed design are being modular, low environmental impact, exclusion of severe accidents, short construction period, flexible adaptation to demand, excellent load following characteristics, and competitive economics.

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INNOVATIVE PWR DESIGN CONCEPT WITH LOW PRIMARY CIRCUIT PRESSURE

D.A. POPOV, B.S. NEYKOV

Department of Thermal and Nuclear Power Engineering
Technical University of Sofia, Sofia, Bulgaria

Abstract. The objective of this paper is to present a concept of a safe and economic pressurized water reactor design with the goal of various applications, primarily large electric power generation, but flexible enough to be used for combined heat and power generation. The essential feature of the proposed concept is the use of a reactor core coolant having very low pressure. In order to minimize the damages caused by of all types of LOCA it is suggested design pressure of the coolant in the primary circuit to be reduced drastically. In this way radical simplification of all the equipments, systems and building structures of the plant could be achieved.

1. INTRODUCTION

For the past 20 years, the commercial nuclear electrical generation market has been void of significant and sustained orders for new plants. In countries such as Japan and Korea, new units have been continuously added to the electrical grid, but these additions have only been adequate to sustain their own country's nuclear infrastructure [1]. Elsewhere around the world, suppliers of new nuclear plants have had to make dramatic shifts in the product portfolio to focus on fuel and service activities only. The main reasons for this negative trend are unavoidable economic factors.

As demonstrated in the accompanying Table I, nuclear power is a far more expensive and risky way of generating electricity than highly efficient combined cycle natural gas plants.

Most of the figures set in Table I are based on the following sources:

— For nuclear plant costs (cases 2 and 3) Koh's study [3] and [4];
— For natural gas cost web source [5];
— For combined-cycle power plant costs [6] and [7];

It is clear that each new nuclear plant built can typically be expected to cost from about $1 billion to several billion dollars in excess lifetime costs. This vast sum of money would have to come in the form of subsidies from governments and/or electricity ratepayers (in the form of higher prices). Obviously it could be much more efficiently used to make investments in energy efficiency, cogeneration, renewables, combined-cycle power plants, fuel cells and the like. As a result of this development, of the next 1000 power plants to be built in the United States, as many as 900 of them are likely to use combined cycle gas turbines and most of the rest will be coal based. The same is situation in Europe. Even in France, which is highly dependent on nuclear power; officials have admitted that combined cycle electricity plants using natural gas are more economical than nuclear power plants [8]. To ensure that nuclear power remains a viable option in meeting energy demands in the near and medium terms, new reactor designs are being developed in a number of countries. Common goals for these new designs are high availability, user-friendly features, competitive economics and compliance with internationally recognised safety objectives. The full spectrum of advanced nuclear power plant designs or concepts covers different types of designs [9]- evolutionary ones, as well as innovative designs that require substantial development efforts as it is shown in Figure 1.

Several evolutionary designs have reached a high degree of maturity: nuclear regulatory authorities have certified some designs, and some are progressing through an optimisation *phase to reduce capital cost*.

Table I. Estimated costs: combined cycle vs. nuclear plants [2]

Power System	Capital Cost $/kW	Interest + Depreciation ¢/kWhe[a]	Nat. gas price $/million Btu[b]	Fuel Cost ¢/kWhe	Non-fuel O&M ¢/kWhe[c]	Total cost ¢/kWhe
Combined cycle Gas Turbine Plant (CC)[d]						
Case 1	500	0.76	1.50	1.02	0.48	2.26
Case 2	500	0.76	2.50	1.71	0.48	2.95
Case 3	500	0.76	4.00	2.275	0.48	3.515
Nuclear Power Plant[e]						
Case 1	1500	2.28	-	0.6	1.7	4.58
Case 2	2500	3.81	-	0.6	1.7	6.11
Case 3 Nuc[f]	4000	6.09	-	0.7	2.0	8.79

[a] Interest and depreciation assumed to be 10 percent in all cases. Capacity factor assumed to be 75 percent in all cases.

[b] Btu stands for British thermal units. 1 Btu = about 1,055 joules. One kWhe (kilowatt-hour electrical) = 3.6 million joules = 3,413 Btu.

[c] Non-fuel nuclear costs include 0.2 cents per kWhe for waste disposal and decommissioning, except in the worst case (case 3) where this cost is taken to be 0.5 cents per kWhe. See Cohn, p. 155.

[d] Efficiency of the combined cycle plant is assumed to be 50 percent in case 1 and 2 and state of the art CC efficiency 60% in case 3. (Nuclear power plant thermal efficiency is about 33 percent. The exact figure does not affect power costs substantially, since fuel costs are a small fraction of total costs.)

[e] Nuclear costs do not include any reprocessing and plutonium management costs.

[f] The worst case capital cost of nuclear (case 3) was typical of US costs for plants coming on line after 1983 but with far higher capacity factor than was typical of the 1980s in the US. The best case nuclear capital cost (case 1) is that reported by the media for sales of Russian VVER-1000 reactors to China.

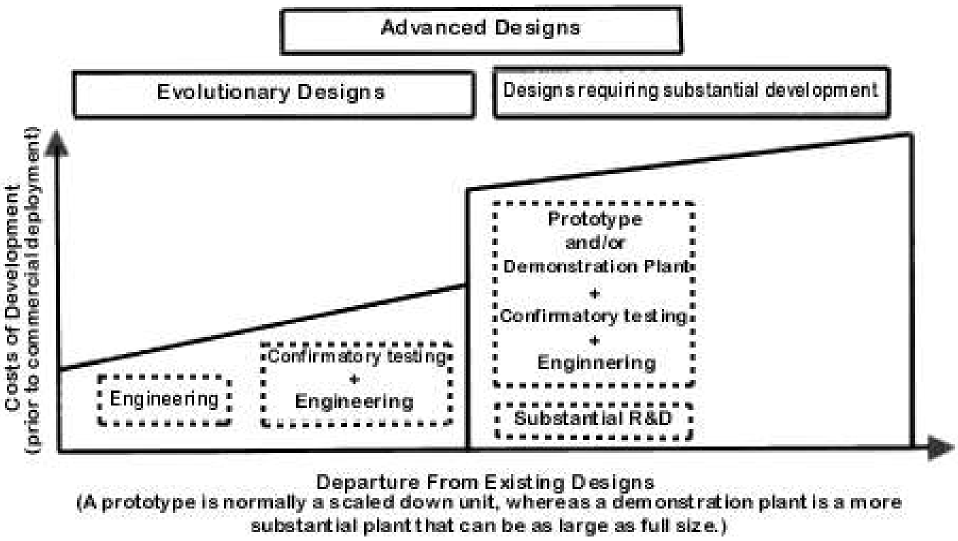

FIG. 1. Efforts and development costs for advanced designs versus departure from existing designs [10].

2. LOW CAPITAL COSTS PWR DESIGN CONCEPT DEVELOPMENT

Ever since electricity began to be generated by nuclear power, the pursuit of cost efficiency with safety as a prime condition has been a great challenge in development work. In nuclear power plants for which plant construction cost is larger compared to fuel, the scale effect has been the ***only*** most effective means of improving cost efficiency, and the capacity of single reactors has been increasing using past results to build upon [11], [12]. At the same time, other efforts have not been paid to ensuring real simplification of the plant and to lowering the huge capital costs.

It is well know that, the design basis accident, for a Pressurized Water Reactor (PWR) nuclear power plant is a large loss of coolant accident (LOCA). In these design basis accidents it is assumed that a break in the piping causes coolant loss and the temperature of the uncooled core increases. Numerous and expensive safety systems are designed such that in the event of such an accident the radioactive release from the core is limited and contained within the containment, a thick concrete dome covering the entire reactor system. All these features made every PWR nuclear power plant very complex and very expensive machine with many parts.

The severity of LOCA depends strongly from the break mass flux. As larger the break and the mass flux are as much severe sequences could be expected. Fig. 2 shows the dependence between the mass flux and the primary circuit pressure. In the range between 16 and 10 MPa the mass flux/pressure dependence is almost proportional. The state of the art PWR plants have got a coolant with a pressure of 15 to 16 MPa. Some old reactors like the Russian made VVER-440 have a primary pressure of 12.5 MPa. As it could be seen in the Fig. 2 the mass flux sharply decreases when the coolant pressure is lower than 8 MPa. This dependence is slightly different when the temperature of the coolant is lowered as well. But it could be firmly stated that any PWR with low primary circuit pressure will experienced far less LOCA sequences than the conventional PWR working with high pressure coolant.

The primary objective in this study was to achieve a plant concept that exceeds significantly all existing PWR concepts in overall cost efficiency and to try to made PWR design competitive with the gas turbine plants and other fossil fuel plants. Based on the above needs, basic design guidelines for a large PWR can be summarized as follows.

— Achieve capital cost efficiency equivalent to that of a fossil fuel plant while maintaining safety, reliability, and operability.
— Design with a high degree of freedom that can respond flexibly to different regional circumstances and investment capacities and to user needs.
— Perform rationalized design that makes use of technologies developed for large-sized reactors.

The heart of the proposed by us concept is the use of a reactor core coolant having very low pressure. In order to minimize the damages caused by of all types of LOCA it is suggested design pressure of the coolant in the primary circuit to be reduced drastically from the current design values of 15-16 MPa up to the values 4-5 MPa. In this way, radical simplification of all equipments, systems and building structures of the plant can be achieved.

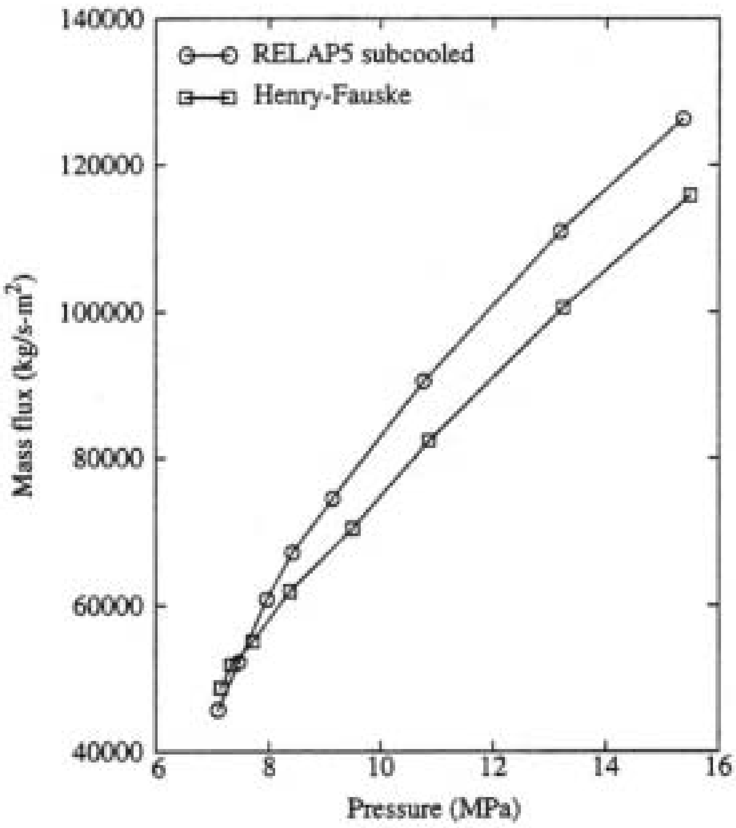

FIG. 2. RELAP 5 subcooled critical flow compared with Henry-Fauske tabulated values in case of different primary circuit pressure [13].

3. LOW PRESSURE DESIGN FEATURES

The features of a plant based on the proposed concept are described below. We were looking for a rationalized design through tradeoff with performance .When the capital costs are huge, the benefit of improved thermal efficiency is relatively small, and for this reason, we placed emphasis on simplifying facilities rather than improving efficiency in the development of a large size evolutionary but as well as innovative PWR design as it incorporates radical conceptual changes in design approaches in comparison with existing practice.

It was calculated that the loss of coolant at lower pressure in the primary circuit is several times lower than the loss when high design pressure is applied. As the pressure in compliance with the temperature in the primary circuit is relatively very low, the evaporation of the coolant during the depressurization of the first loop will be minimal. It was estimated that while reducing the design values of the pressure from 15-16 MPa to 4-5 MPa, the accidental evaporation of the coolant will be several times lower.

The presented at Fig. 3 curves show the benefits coming from coolant pressure lowering. It is obvious that when the primary circuit pressure is set at its lower level the amount of released vapors will be minimal. As a result the containment vessel pressure will be also minimal. The aforementioned curves are calculated using simple thermodynamic assumptions and equations [14]. Furthermore, Fig. 3 is based on large and constant containment size/volume typical for existing widely spread PWR plants with single capacity of 1000 MW and more. In case of new low primary circuit pressure PWR plant, the volume of the containment could be reduced due to the minimal break mass flux and minimal reactor coolant evaporation.

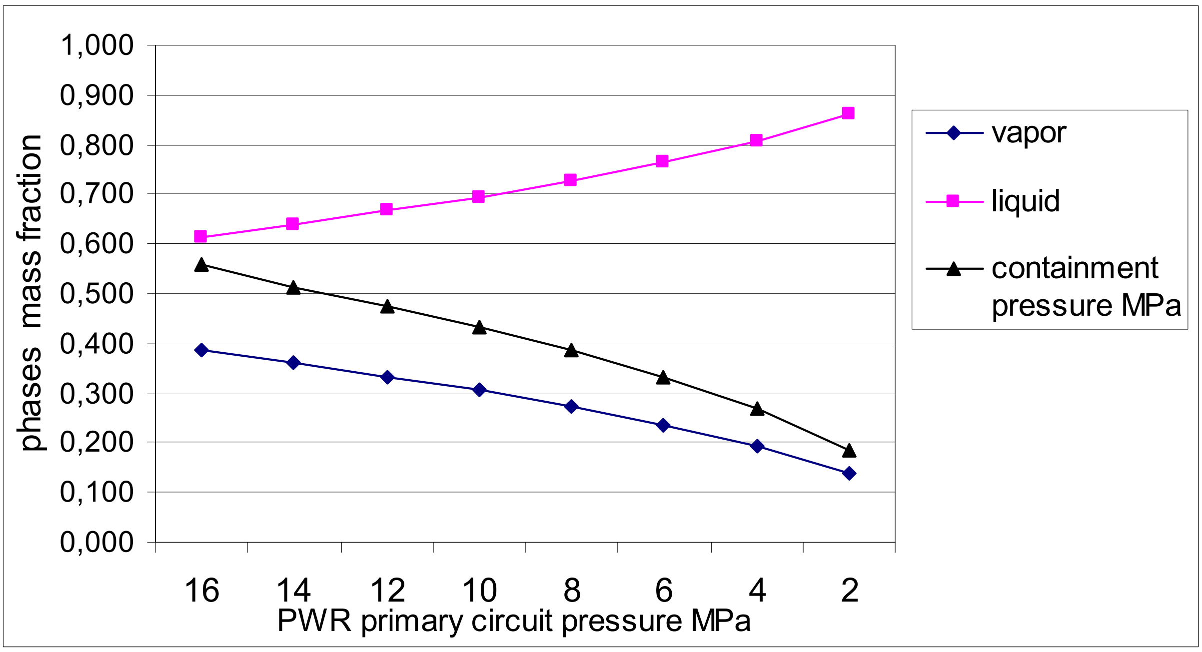

FIG. 3. Liquid and vapor mass fraction development after sudden depressurization of PWR coolant.

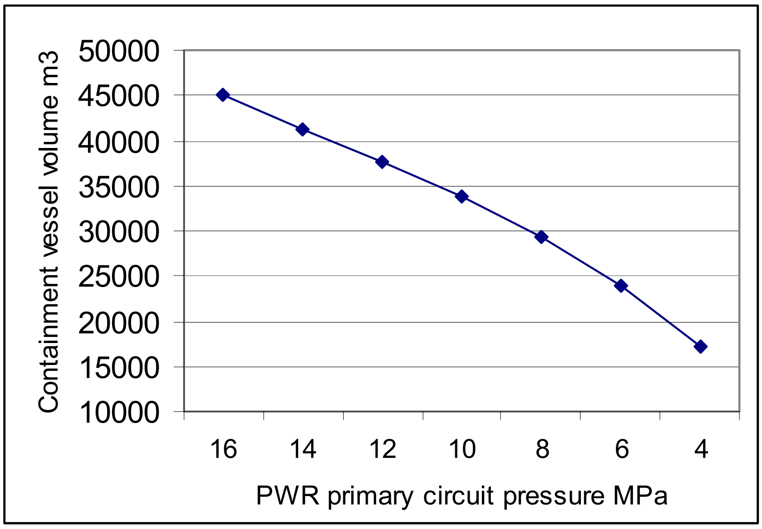

FIG. 4. Containment vessel volume reduction in case of low pressure PWR design for different primary circuit pressure (containment pressure fixed at 0,56 MPa for all cases).

The presented at Figure 4 curve demonstrates containment vessel volume reduction in case of low pressure PWR design when only reactor coolant evaporation is considered as a factor determining containment pressure. So, it is realistically to expect that PWR with reactor coolant pressure of 4 MPa might have 3 times lower containment volume (or smaller enough) than the existing PWR plants.

Figure 5 presents a comparison between the containment structure of high pressure and low pressure PWR plant. The maximum containment pressure is fixed at equal values for both cases. Obviously the construction cost can be reduced significantly when the low primary circuit pressure is applied. Therefore large capital cost savings could be achieved as a result of the proposed innovative PWR design.

The evaporation of the coolant during depressurization of the first loop has been also estimated in the case of different containment volumes at fixed internal pressure.

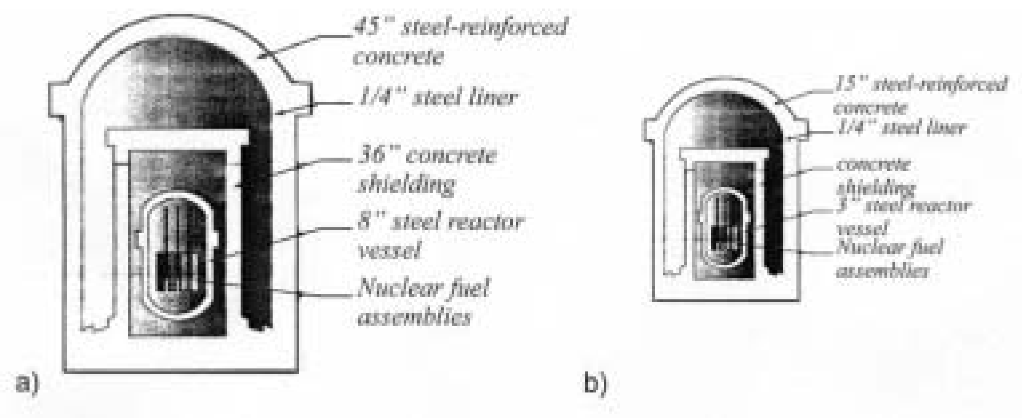

FIG. 5. PWR Containment size reduction: case a) - typical large PWR containment for USA nuclear power plant [15]; case b) – containment of PWR with low primary circuit pressure.

The main figures are presented at Figure 6. Only very small portion of the coolant will be evaporated in the case of low primary circuit pressure. As a result, an improvement in cooling the reactor core could be apparent and all kinds of accidents will be alleviated. This approach will give an opportunity to reduce the number and at the same time to simplify all emergency cooling systems. In this way furthers capital costs reduction could be achieved.

When we assume reduced parameters in the primary circuit, the steam produced from the steam generators will have very low pressure. This approach creates the possibilities for great simplification of the secondary circuit of the plant. The installation of high-pressure turbine will not be necessary as well as separators, reheaters, high-pressure heaters and all the accompanying pipelines, valves, fittings etc. As a result the size of turbine hall could be reduced twice at least.

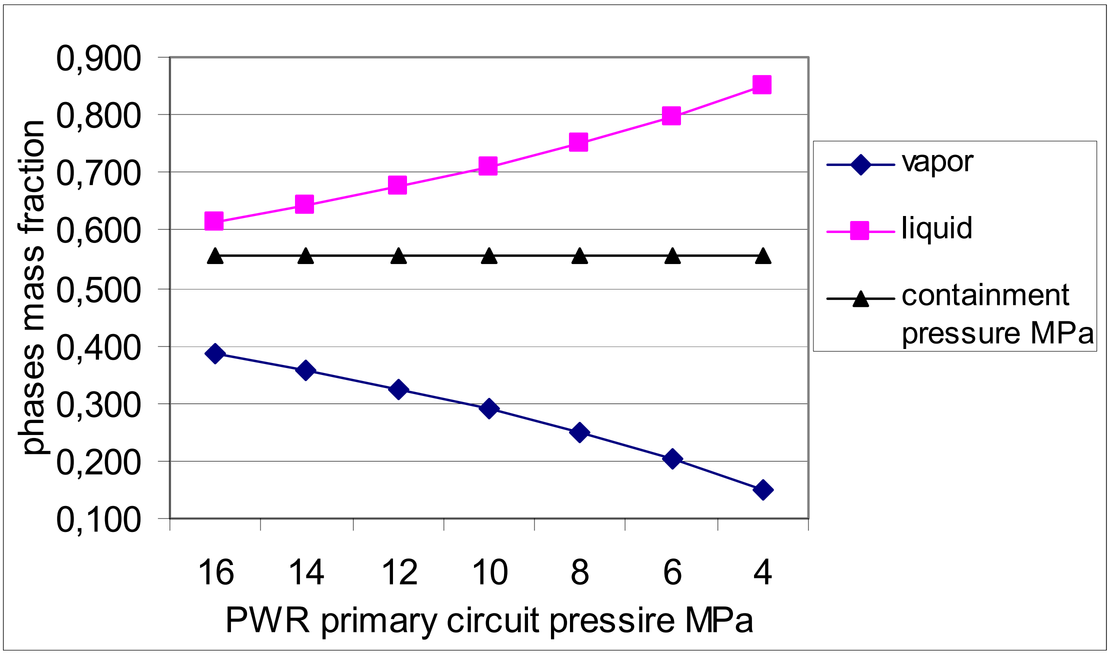

FIG. 6. Liquid and vapor mass fraction development after sudden depressurization of PWR coolant in case of containment vessel volume lowering (containment pressure fixed at 0,56 MPa for all cases).

4. IMPACT ON FUEL CYCLE AND POWER PLANT ECONOMY

In the presence of lower values of live steam pressure, the heat rate of the second loop deteriorates. It was evaluated that the pressure reduction in the first loop from values of 15-16 Mpa to values of 4-5MPa, will lead to efficiency drop from 33% to 24%, Figure7.

However, because of the negative thermal reactivity coefficient and excellent neutron economy, the reactors with low parameters of the coolant will have a longer fuel cycle while we have the same degree of fuel enrichment of U^{235} compared to the reactors with higher design parameters [16]. This will be the reason for increased fuel burn-up and the fuel will be used more effectively.

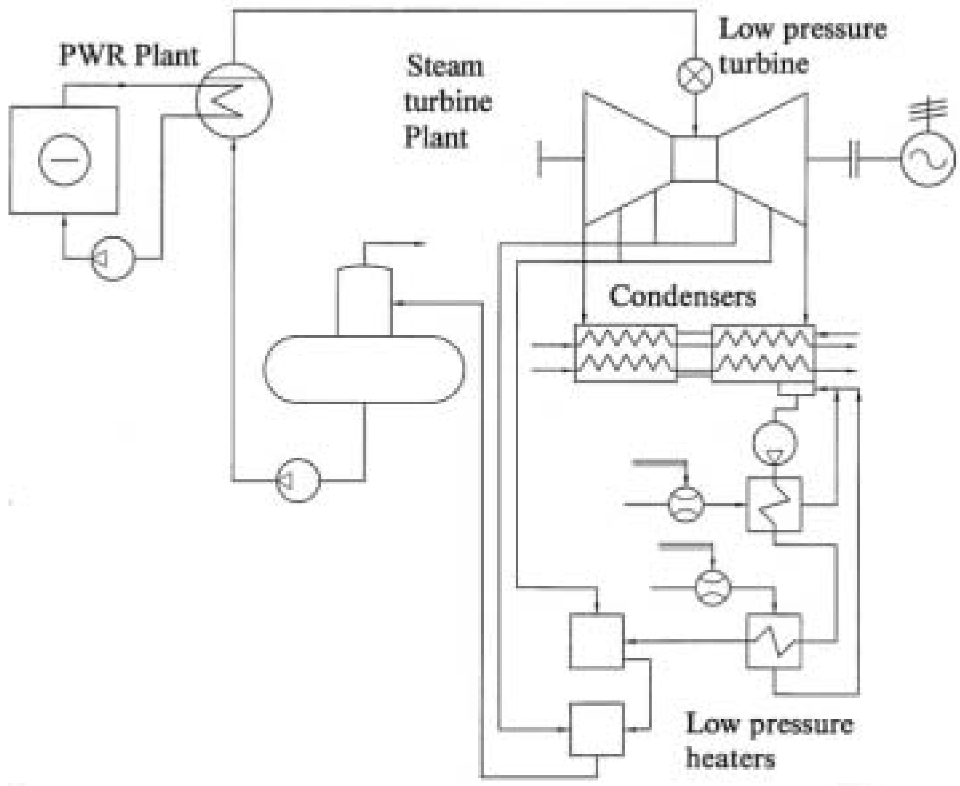

FIG. 7. Low pressure PWR plant with simplified secondary circuit design.

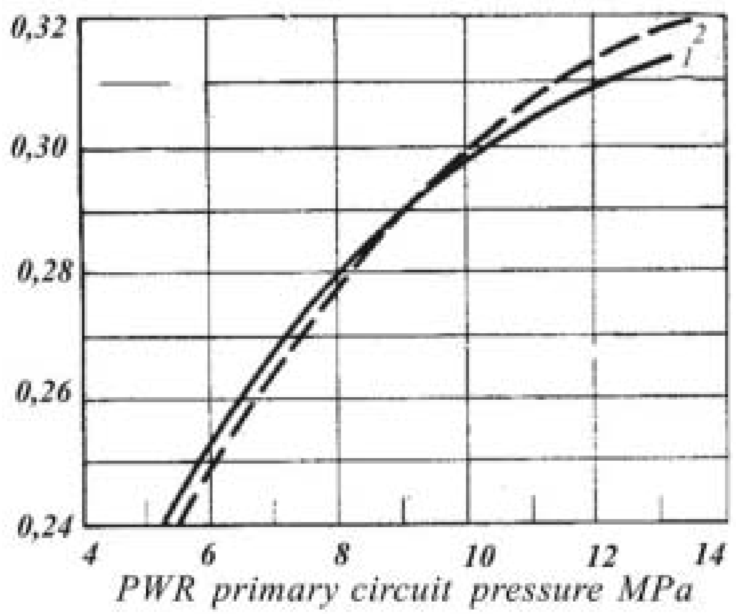

FIG 8. Thermal efficiency of PWR plants with different primary circuit pressures: case 1-without steam reheating; case 2-with steam reheating.

Another way for bettering low pressure PWR plant economy is to develop a new fuel cycle similar to DUPIC cycle [17]. Depending upon initial enrichment and burn-up, spent LWR fuel nowadays contains about 0.9 wt% U-235 and 0.6% fissile plutonium. It is a somewhat higher value at lower LWR burn-ups, and it can vary by a few percent between 40 MWd/kgU and 50 MWd/kgU. This level of residual fissile enrichment represents more than two times that of natural uranium (0.71%). The predominant nuclear reactor types in the current world market are LWR and this trend will likely continue for the foreseeable future. Under such perspective, the question of "how to manage the spent fuel discharged from those reactors" will remain as a key issue to be considered, for the sustainable supply of nuclear energy in the future. The conventional backend fuel cycle has evolved into two different directions, depending upon national policies: either direct disposal in deep geological formations, or to reprocessing of spent LWR fuel for MOX fuel recycle in LWR (or FBR). However, the decision on which of these two options to pursue is still pending for many other countries with "wait and see" position.

The proposed in this study innovative PWR design concept could offer an alternative fuel cycle to those conventional options when looking at the possibility of reusing the spent LWR fuel again in PWR with low primary circuit pressure, by taking advantage of the inherent high neutron economy. The spent LWR fuel could be transformed into "fresh" fuel by direct re-fabrication, without any separation of nuclear materials. According to physics calculations done at TU-Sofia, for a fuel with this residual fissile content, the initial effective multiplication factor k_{eff} is high enough for good discharge burn-up in PWR with primary pressure below 6 MPa. For an average conventional LWR burn-up of 45 MWd/kgU, an additional 10 MWd/kgU can be expected using the fuel cycles synergism, or 22% more power than the once-through case.In this way, the fuel cycle economy of the nuclear power plant with low parameters could be comparable to the current PWR plants with high parameters.

5. CONCLUSIONS

The preliminary analyses that were carried out showed that the following significant advantages of the nuclear plants with low design parameters in comparison with the state of the art PWR plants with evolutionary design could be achieved:

— a very simple design to expedite licensing, reduce capital cost and reduce construction time,
— a simpler and more rugged design, making them easier to operate and less vulnerable to operational upsets,
— higher availability and longer operating life,
— reduced possibility of core melt accidents,
— minimal effect on the environment,
— higher burn-up to reduce fuel use and the amount of waste
— unique fuel cycle synergism with the existing LWR technology

The proposed concept deserves further intensive research and development works in order to prove its vitality.

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NUCLEAR ENERGY FOR OIL SANDS PRODUCTION: PROVIDING SECURITY OF ENERGY AND HYDROGEN SUPPLY AT ECONOMIC COST

R.B. DUFFEY, A.M. MILLER, J.M. HOPWOOD, S. KURAN, H. KEIL, D. BOCK

Atomic Energy of Canada Ltd
Mississauga, Canada

Abstract. The development of Canada's 2000 EJ oil sands resource depends on a substantial energy input for extraction and upgrading. So far, this input has been supplied by natural gas, a resource that (a) is a premium fuel; (b) has limited availability; and (c) produces significant CO_2 emissions. For the now preferred SAGD *in-situ* method of extracting oil-sands bitumen, nuclear heat can easily supply the steam at the ~ 2.5 MPa requisite pressure. Studies by AECL and others show that steam from an Advanced CANDU™ Reactor (ACR™) should produce steam for SAGD at lower cost than natural gas and also give far greater price stability. The large quantity of steam (2 to 2.5 volumes of condensate per volume of bitumen) for a typical project of 100 to 140 million barrels per day of bitumen provides a good match to the output of a 1900 MW(th) reactor, which would also produce about 200 MW of electricity. Electricity would be produced using a back-pressure turbine, yielding a very high overall energy efficiency. AECL work also shows economic competitiveness for electrolytic production of hydrogen, which is needed to upgrade the bitumen. Electrolysis would be interruptible, avoiding the short periods of high electricity prices experienced on the Alberta grid. Competitiveness with conventional steam-methane reforming is achieved by a combination of off-peak power and low-cost electrolytic cells. Using nuclear-generated steam and electricity produces negligible CO_2, thus placing synthetic crude from the oil sands on a comparable basis to conventional crude with respect to greenhouse gas emissions.

1. INTRODUCTION AND BACKGROUND

The approximately 12% of Canada's Oil sands reserves that are estimated to be recoverable exceed 300 billion bbl versus a world production of approximately 25 billion bbl/a. Overall, Alberta's conventional hydrocarbon reserves were estimated in 1995 as ~ 140 EJ of natural gas, about 40 EJ of coal*, and over >2000 EJ of oilsands bitumen [1]. Since the SAGD recovery potential is approximately 80% as a fraction of the Oil Sands reserves, to extract and upgrade all the bitumen available for SAGD just in the Fort McMurray area, would take 60% of remaining Alberta natural gas [2], and more than 100% of remaining established reserves of Alberta coal.

Therefore, AECL's studies have examined the potential benefits of applying CANDU energy to extending and supporting oil sands extraction, processing and upgrading. The work was also prompted by recognition that the evolution of the hydrocarbon market may open up a competitive advantage for CANDU and associated technologies to supply energy for SAGD-based oil sands projects. In particular, the study examined the potential benefits of stable energy and H_2 prices, at levels less than projected alternatives. In addition, as an independent

* This is the "established coal reserves". Ultimate Alberta reserves are estimated to be about 20 times larger but 75% is estimated to be low-grade sub-bituminous and accessibility beyond the "established" reserves is unestablished.

body, the Canadian Energy Research Institute (CERI) was contracted to compare the economics of nuclear and gas-fired options to supply steam to an oil sands reservoir using SAGD.

The objectives were to compare the economics of a modified Advanced CANDU Reactor (ACR) 700 with a gas-fired cogeneration facility, to supply steam to a hypothetical SAGD project located in north-eastern Alberta. The comparison was at a pre-feasibility level, and the scope of the studies included the provision of a CANDU reactor to supply steam and co-generate electricity, and the use of electrolysis for the production of hydrogen and synergistic applications of by-product oxygen and heavy water production. The study covered key issues of technical applicability, economics and schedule. The initial results of the studies are reported in this paper.

The scope of this Initial Feasibility Review was to establish the economic potential, and to confirm the feasibility of a CANDU-based energy supply system for SAGD oil production. The CANDU nuclear power plant system is a well-understood energy source from applications elsewhere. Therefore this review has focused on the feasibility and economics of the specific application of CANDU and associated technologies to an oil sands project.

The scope of the study includes the following:

a) Steam supply from a nuclear (CANDU ACR) plant for SAGD oil extraction.
b) Co-generation of electricity from nuclear to supply project and other commercial uses.
c) Hydrogen production from electricity by electrolysis, for use in upgrading the extracted bitumen.
d) Use of the oxygen by-product of electrolysis to enhance sulphur recovery.
e) Side-stream use of electrolytic hydrogen to extract heavy water for sale to CANDU customers.

With reference to (a) and (b) above, the scope of implementation is fundamental to overall development planning. Therefore we cover all aspects of feasibility, specifically technical issues, comparative economics, schedule, regulatory requirements, and other implementation factors.

With reference to items (c), (d) and (e), the application to oil sands operations does not involve any significant changes from available technology, and schedule requirements are readily met. Thus this review focuses on comparative economics of the latest applications of CANDU technology, and on current projections of schedules and economic parameters. The review also builds on and updates information and insights from previous studies, in particular the 1994 AOSTRA study of CANDU application to oil sands extraction [2, 3].

The Energy Supply is based on the ACR adaptation of AECL's medium-sized CANDU 6 nuclear power plant. Ten units of the CANDU 6 nuclear power plant are in operation, or under construction in Canada and in several countries, most recently in 2003 in China. A technical outline of the ACR plant is given elsewhere [3]. The ACR is an evolutionary improvement on the CANDU plant, with:

— technology upgrades to substantially reduce specific capital cost;
— increased steam supply pressure and temperature;
— further reduction to the already low environmental emissions and waste production;
— reduced construction cost and schedule; and
— streamlined operations.

Since the ACR product offers a significant cost advantage, it is proposed as the most economically attractive option for energy supply to oil sands project operation. For completeness, we have also considered the costs and potential of new generation IV reactor designs (specifically the VHTR and the SCWR systems) for hydrogen production.

2. STEAM AND HYDROGEN - EVALUATING THE COST

One key output is Supply Cost, being the constant-dollar price that would recover all costs including Capital, Operating costs and Return on investment. Supply cost is calculated using discounted cash flow techniques and, since AECL is non-taxable, comparison supply costs for both the nuclear and the gas-fired options have been calculated before tax.

Steam Assist Gravity Drainage (SAGD) is used for *in-situ* extraction, with a series of stages in the range of approximately 35,000-70,000 bbl/day bitumen each, developed in coordination with expansion of the oil upgrader capability.

2.1. Energy requirements and forward price stability.

The energy supply for oil sands production is a very significant element in the overall production process. Relative to the total energy value of the final synthetic crude oil (SCO) product, approximately 18% is used in extraction via SAGD, with a further 10% or more used in upgrading. Therefore the characteristics of the selected energy supply option have a very large effect on the characteristics of the overall project, with regard to sustainable development principles.

SAGD is an enhanced oil recovery process applicable to *in-situ* recovery of crude bitumen from deep oil sands deposits. A typical application involves twin horizontal wells drilled in parallel, with one a few metres above the other. During the start-up phase, medium pressure steam is circulated in both wells to heat the reservoir of bitumen-sand mixture by conduction. The heating reduces the viscosity of the bitumen, increases its mobility, and establishes pressure communication between the two wells along their length, so that a flow of fluids can occur from the upper well to the lower well.

Once communication is established, the lower well is placed on production and the upper well injects steam, representing the start of “normal” SAGD operations. Continued steam injection gradually creates a steam chamber above the well-pair, which expands upwards to the top of the reservoir and laterally until contact is made with similar steam chambers from adjacent well-pairs.

The steam injection rate through the upper well is increased until the desired reservoir operating pressure is achieved, then varied to maintain that pressure. Driven by this pressure and by gravity, liquid bitumen and water flow as a mixture from the upper well to the lower well. The mixture then flows to the surface where it is processed to recover the bitumen and recycle the water through cleanup plant and closed-cycle boilers. The flow rate from the production well (lower well) is controlled so that the “bottom hole” (in-well) temperature of the produced fluids is several degrees Celsius below the saturation steam temperature at the operating pressure. This prevents steam from breaking through into the production well.

A typical nuclear system would provide the 60 to 80,000 m^3/d steam capacity (stream-day) of 80% quality steam for the pressure range (~ 2.2 to 3 MPa) required for SAGD extraction.

A range of parameters have been considered, with a typical reference project output of 62,400 m^3/d of 100% quality steam with an assumed cumulative steam/oil ratio of 2.5 for a hypothetical SAGD project size. For an operating capacity of 93%, the result is consistent with a 23,200 m^3/d (146,000 bbl/d) bitumen production SAGD operation in northwestern Alberta, targeting a high-quality Athabasca resource.

A project of this size is large relative to existing commercial SAGD projects; however, several companies are proceeding with projects in the 12,700 m^3/d (80,000 bbl/d) to 15,900 m^3/d (100,000 bbl/d) range and beyond. For example, one of the largest projects envisages a series of stages of 35,000 bbl/day or 50,000 bbl/day bitumen each. With four to eight stages expected, this gives an ultimate potential output of 140,000 to more than, 300,000 bbl/day. (The first stage of this project is expected to start production this year.)

Note, for the initial studies, the plant gate price of natural gas is taken as Can$3.50/million Btu (3.32 $/GJ) and electricity at Can$45/MWh. The most recent study with CERI assumed a gas price of Can$3.50/million Btu and an electricity price of Can$50/MWh, similar parameter values to the earlier work. Recently, natural gas prices have fluctuated widely, up to double this reference level, whereas nuclear electricity production prices are and have remained stable (i.e., fuel insensitive). Therefore, the nuclear energy option allows an oil sands project to secure its source, and decouple its energy use prices, rates and costs from the market variations caused by natural gas supply and demand, and can also project forward confidently without large uncertainty and risk in energy prices.

Steam generation is the main user of energy in the production of bitumen with the SAGD approach. However, significant energy is also consumed to upgrade the bitumen to synthetic crude oil. The reference planning basis currently assumes upgrading by hydro-treating, which would use about 3 kg of H_2 per barrel of bitumen. If this were produced by a Steam Methane Reformer (SMR), the natural gas requirement would require a further 50% of natural gas beyond that used for steam production.

2.2. Hydrogen production

Since the process requires hydrogen at a rate of several kilograms of hydrogen per barrel of bitumen, we have reviewed the alternative way of producing hydrogen, via electrolysis of water, with CANDU electricity as the energy source. The reference economic comparison is of electrolytic hydrogen versus SMR-generated hydrogen. However, the use of electrolysis also enables two additional opportunities: electrolysis also produces oxygen, which can be used to enhance the capacity of any sulphur recovery plants; also, using Combined Electrolysis and Catalytic Exchange (CECE) technology, the electrolysis plant can be readily adapted to produce a side-stream of heavy water (D_2O), an essential component of CANDU technology. The economic impacts of these two options have been estimated.

In addition, we have examined so-called Generation IV nuclear reactor options (VHTR and SCWR) for hydrogen production to establish the potential of these concepts. These are presumed to adopt the so-called Sulfur-Iodine (S-I) Process or some similar way of dissociating water at temperatures in excess of 850°C [4].

While the Nuclear plant (ACR) will produce electricity in co-generation with the medium pressure steam used to extract the bitumen, the use of electrolysis to produce hydrogen is not directly linked to the unit. Steam is required at the SAGD site; hydrogen is consumed at the main processing site. For this reason the economics of each of these energy alternatives is assessed independently in this report. With regard to credit for CO_2 offsets, the credit for zero-emissions electricity production is associated with the CANDU unit.

The analysis has to consider fluctuations in the price of electricity characteristic of the local power pool. It may pay for the project to buy or import power from the grid (when electricity is cheap), and export (sell) to the grid when the price is higher, at times of peak demand. We have studied the actual market price variations to determine the beneficial effect this sales opportunity may have on both profitability and on plant design. Basically, when electricity is cheap it is economic to produce hydrogen, but above a threshold pool or market price, it is better to sell electricity and draw down storage (see Figure 1). Since the supply of hydrogen must be maintained, interruption of production must be offset by sufficient hydrogen storage and by a higher instantaneous rate of production. An alternative to storage would be turn to a reformer to expand its hydrogen production.

If electrolytic hydrogen production were of a sufficient scale (around 100 MW or 19 million scfd or greater), the addition of a Combined Electrolysis and Catalytic Exchange (CECE) plant to produce heavy water (D_2O) would likely be profitable. Electrolysis would also produce oxygen.

Traditionally, SMR technology for large-scale H_2 production has had an advantage of lower energy and lower capital costs. However, new cell technology is intended to lower the fully installed capital cost to around ~170 \$/kW [5] which would bring electrolysis to below the capital cost of an installed SMR. However, operating costs are likely to be higher for electrolysis since ~ 50 kWh is needed to produce each ~1 kg hydrogen for the cell efficiency typical of these low-cost cells. Note though that the use of electrolysis may have an additional economic value by providing a demand for excess electrical capacity in the local grid or power distribution area, reducing or eliminating potential tie-line costs.

At the outset of assessing nuclear plants for the oil sands, the opportunity for substantial cost-savings from co-locating the nuclear plant and an electrolytic plant was noted (e.g., generation of DC power directly). This is now appreciated to be marginal since any upgrading will likely take place in at the main oil sands site, remote from the nuclear plant.

Hydrogen production required for upgrading is a commercial matter and not public information. However, we estimate that for upgrading bitumen, perhaps up to some 70 million scfd could be needed. If produced electrolytically, this requires 368 MW, so the electricity demand is of the right order to match the production of hydrogen for upgrading by electrolysis. However, buying extra power or using only some of the CANDU-generated electricity output in this way would be perfectly practicable. In addition this amount of hydrogen is quite within the reach of the heat and power produced by advanced designs like the SCWR and VHTR, and for the SCWR could be derived directly from a "topping" heat cycle, while also producing electricity and process heat.

The effect of any assumptions on capital cost, CH_4 cost, and CO_2 credit can be pro-rated. The actual cost of electricity to an electrolysis unit can be notably lower than the average pool price if the electrolytic hydrogen supply is configured to take advantage of off-peak electricity by storing H_2 for use at times of high electricity price. Electrolytic H_2 is often promoted as a way to load-level electricity demand, with electrolytic power diverted to the grid during periods of peak price/demand. The additional electrolytic capacity has a modest cost but storage to provide uninterrupted H_2 supply could be a substantial factor – a recent study by the Joint Institute for Energy and Environment [6] uses 1 US\$/ft^3 for H_2 storage.

Figure 1 applies our pricing model to the actual hourly Alberta Pool Price Data for 2002. The Pool buying price averaged just under 30 US$/MW.h but the average conceals very large variations, with very high-price spikes above a fairly low-cost norm. Beyond a concentration in periods of cold winter weather, the price spikes did not have an easily discerned pattern but the pattern is predictable, the Pool declaring the price one week in advance. Apart from storage, interruptible electrolysis requires additional electrolytic capacity to offset the idle periods. One can analyse the real electricity cost data for a range of electrolytic excess and storage and adjust these two parameters so that hydrogen is always available from storage.

For this large-scale application, cells are costed at 170 US$/kW while consuming electricity at 2.1 volts (including power to compress the product gas).

Figure 1 shows a flatish, minimum cost for hydrogen around 1700 US$/t where electrolysis is confined to periods when the grid price for power is less than 55 or 60 US$/MW.h. The *average* electricity price for almost 95% of the year was only 22.4 US$/MW.h when the cut-off was 60 US$/MW.h. (For the remaining 5% of the year, the price averaged 157.8 US$/MW.h.). Excess electrolysis capacity (above that for the continuously operated option) was around 20 to 25% and about 16 hours of storage would have sufficed.

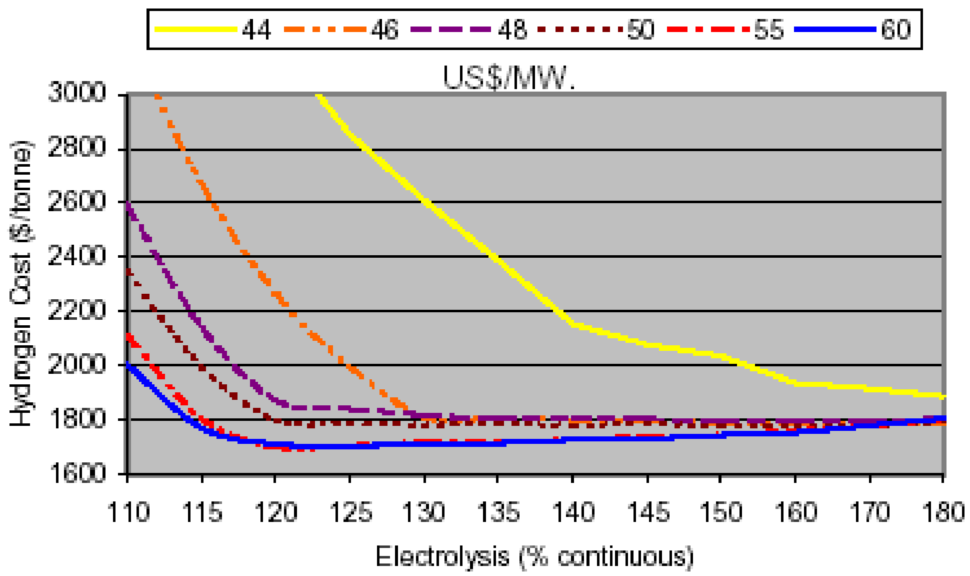

FIG. 1. Cost of Hydrogen as a function of the threshold cost of electricity (44 to 65 $MW.h) above which is not produced and as a function of electrolytic capacity relative to that required for continuous production.

Interruptability of process inputs is important but an issue of contrasts for oil sands processing. The SAGD process operates in a timeframe of weeks and months so the only effect of quite extended interruptions to steam supply would be a reduction in average bitumen flow over an even longer period. By the standards of most chemical plant operations, SAGD is most unusually tolerant of steam interruption because the product flow responds only slowly. In sharp contrast, the processes of bitumen upgrading would be severely disrupted by the briefest of H_2 supply outages. Storage would be one approach to maintaining H_2 supply if H_2 production were to be switched off when the value of electricity is high. However, the quantities of H_2 to be stored would be large, expensive, and likely dependent on the use of underground caverns. An alternative approach would be to use electrolysis to produce only around half of the total H_2 stream. While SMRs cannot rapidly be brought on-stream from cold, the output of an operating SMR can be varied quite rapidly, at least between 50 to 100%. So shut-downs of electrolytic H_2 could be offset by raising the production of an SMR from normal operation near 50% of capacity to 100% capacity. The predictability of electricity prices on the Alberta Pool grid would further facilitate this approach of output variation from an SMR source.

3. ECONOMIC BENCHMARK AND ASSUMPTIONS

In the competitive energy and power business, nuclear energy must be reviewed as an alternative to the reference energy supply, in this case natural gas. Thus, it would only be considered for implementation (after further study) if shown to have an economic advantage, and if non-economic implementation issues can be satisfactorily addressed.

The financial ground rules and assumptions for AECL's initial study were therefore based on a competitive and commercial basis:

— Assume projects are financed by a private consortium, hence commercial value of the IRR is expected, based on a total-project calculation.
— Assume all costs and revenues escalate by a uniform 2% per annum.
— Assume current reference rate for electricity revenue/cost (~$45/MWh).
— Assumed plant-gate natural gas cost ($3.5/million Btu).
— CO_2 credits are only accounted as a sensitivity (i.e., not built into the costing).
— NO_x, SO_x and VOC credits are only assumed if emissions are constrained in the local airshed.

4. ENERGY SUPPLY DESCRIPTION

4.1. Nuclear steam and electricity supply

The base-case configuration is based on an initial adaptation of the ACR Nuclear Steam Supply System with the results shown in Table I.

This ACR has been developed from AECL's proven CANDU 6 plant to reduce construction schedule, improve operating performance and further improve environmental performance. Consequently, competitiveness is significantly enhanced.

For reference, the ACR design uses the following key features:

— Compact reactor core consisting of 284 horizontal fuel channels, each containing 12 fuel bundles.
— CANFLEX fuel bundles with slightly enriched UO_2.

— Double-ended, on-power refuelling of fuel channels based on two-bundle per channel replacement for one refuelling operation ("two-bundle" refuelling), using two fuelling machines.
— Indirect-cycle cooling with separate reactor and steam-supply loops.
— Single-loop reactor coolant system with two coolant pumps and two steam generators.
— Passive containment and radiation shielding of nuclear systems by pre-stressed cylindrical reactor building with hemispherical dome.
— Automatic computer control of all reactor and power systems from cold standby to full power operation, by the use of fibre optic-linked distributed control system.

The ACR incorporates improved fuel channel design and the use of slightly enriched uranium fuel, which allows reactor coolant system operation at an increased pressure of 12 MPa. This in turn enables the CANDU steam generators to operate at a steam supply pressure of 6.5 MPa. For reactor application purely for electricity production, this has an advantage of increased overall turbine efficiency. For application of oil sands steam supply, this has a more immediate advantage, enabling all the steam supply to be passed through a high-pressure steam turbine in series with, and upstream of, steam flow to the oil sands boilers. Figure 2 shows a schematic of the concept.

Table I. Heat balance for nuclear options

	Option 1	**Option 2**
Total CANDU Energy Output	1,900 MW(th)	1,900 MW(th)
Energy to Oil Sands Boiler	1,433 MW(th)	1,712 MW(th)
Energy to High Pressure Turbine	188 MW(e)	188 MW(e)
Energy to Low Pressure Turbine	62 MW(e)	Nil
Energy to Condenser	217 MW(th)	Nil
House load for CANDU	40 MW(th)	35 MW(e)
Net Electrical Output	210 MW(e)	153 MW(e)

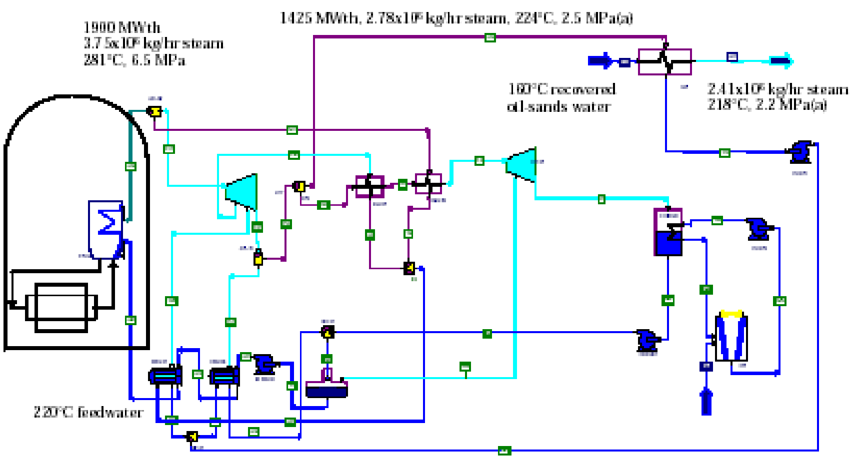

FIG. 2. ACR for oil sands applications.

4.2. *Assured schedule and reduced project risk*

In comparison to previous projects, a next generation CANDU build project would be completed in a much shorter schedule. The first unit of the recent Qinshan Phase 3 project was completed in 71 months from the contract effective date to commercial operation in 2003, one month ahead of schedule. The project schedule for an ACR oil sands plant is estimated at less than 60 months, based on the following improvements:

— The use of light water coolant reduces the time to commission major coolant and auxiliary systems.
— The reduced number of coolant system components reduces manufacturing and installation time for major critical path equipment.
— The reduced number of fuel channels in the reactor core reduces the duration of major reactor manufacturing and installation steps.
— Overall application of modularization technologies for system design, fabrication, and construction, shortens the on site labour requirements, thus time and cost.

The operating cost is expected to be less than 1¢/kWh (equivalent electrical output, excluding fuel costs). This has been already achieved at the latest Canadian CANDUs, at the multi-unit Darlington station.

Relative to the usual oil sands energy supply option, nuclear is low in operating and fuel cost, but is capital-intensive. Therefore for risk comparison, the most important factors are CANDU capital cost risk versus natural gas fuel risk.

At this stage, all capital cost estimates for such a new adaptation are considered preliminary-conceptual, and a typical uncertainty of ± 20% may be appropriate to take into account in estimating capital costs at this time. Responsibility for capital cost risk will be defined as part of the project implementation definition. By comparison, natural gas fuel costs, (which have a similar economic impact to CANDU capital costs) may be subject to much greater uncertainty than 20%, particular in the upward direction. For example, a commonly used “high” estimate for natural gas prices looking ahead is US$6/GJ, typical of prices in recent years, and based on additional exploration, operating and pipeline costs associated with extraction and delivery of arctic gas. Therefore, a high-side cost risk for fuel costs for natural-gas-generated steam may be considered actually at 100% or more.

As to schedule risk, recent CANDU units all achieved full-power operation exactly on schedule. This was accomplished while working with several new equipment suppliers and construction companies who had not participated in a CANDU project (to accommodate local content requirements). This gives confidence that a CANDU project in a new location can be delivered on schedule. Schedule risk for a nuclear energy supply can therefore be considered similar to other large-scale energy projects. That is, an IRR of similar value to other energy projects would represent a similar coverage of the economic impacts of schedule delays.

One schedule risk to be considered for a nuclear project is the potential for of licensing delays, particularly for a new site. However, no delays in obtaining nuclear licenses or permits were experienced for the Wolsong projects, or in obtaining site license and construction permit for the current Qinshan projects. This demonstrates the maturity of the CANDU licensing process. For the CANDU, all licensing activities other than project verification will be concentrated into the Project Development stage, so that licensing risk will have been minimized prior to project start.

Performance risks exist, and the oil sands production facility is intended to operate at a high target capacity factor of > 90%. The CANDU energy supply system would be designed and operated to perform consistently with this target.

5. ENVIRONMENTAL BENEFITS AND PERFORMANCE

CANDU plants offer important benefits in the elimination of conventional emissions to the environment. The nuclear power systems generate and release zero quantities of sulphur oxides, nitrogen oxides and greenhouse gases, and emissions of other environmentally active materials such as oils, solvents and other chemicals, are at negligible levels. Routine emissions of radioactivity to air and water are a small fraction of internationally accepted limits. Resulting radiation doses, even to the most exposed member of the public, are a tiny fraction of everyday background radiation. CANDU reactors have performed consistently with this overall record. In Canada, operating CANDU reactors limit emissions so the dose to an individual permanently located at the plant boundary is less than 1% of the regulatory dose limit, which is itself set far below the typical background and medical dose to individuals in everyday life. To illustrate the low values of radioactive emissions from a nuclear power plant, studies have shown that the radioactive trace elements emissions from coal plants are typically 100 times larger than from equivalent size nuclear power plants [7].

The technology in the ACR plants is aimed at reducing environmental impact further from these levels. The inclusion of new ACR technology also simplifies waste management processes. Waste generated by a CANDU power plant consists of minimal quantities of conventional wastes (lubricating oils, water treatment and sewage plant wastes, etc.), low level radioactive wastes (solid and liquid) and spent nuclear fuel. Low-level solid radioactive wastes consist largely of items such as clothing, rags, and other consumables used in nuclear maintenance. This waste is small in volume and can be stored on or off site in drums and concrete-lined containers. Liquid radioactive waste, largely ion-exchange resin slurry, can also be stored on site. For CANDUs, the option of treating this waste via dewatering and cementation, followed by transportation off-site for immediate disposal, offers the opportunity for cost reduction as well as elimination of on-site waste management responsibility.

For ACRs, the use of slightly enriched uranium fuel allows each fuel bundle to achieve on average three times the energy production of natural uranium fuel used in current CANDUs. This means that spent fuel production per year is only one-third that of natural uranium fuelled CANDUs or about 30 tonnes/a, a modest volume, which is normally stored in a water pool on site for up to ten years. After storage in the water pool, spent fuel can be transferred to passive dry concrete storage canisters, which require no maintenance or other services, and which have a life of many decades. Spent fuel can be safely stored in this way indefinitely until it is convenient to send it for final disposal to a permanent facility or for recycling. The option of transporting spent fuel and low-level radioactive waste off site for long-term storage is expected to be available for an oil sands CANDU. This would eliminate all but temporary radioactive waste storage requirements at the site.

6. TYPICAL THERMAL CYCLE PLANT CONFIGURATION

A typical configuration has the interface between the CANDU steam supply and the oil-sands steam delivery and water recovery/treatment system is defined very simply. The CANDU unit delivers approximately 1,150 MW(th) via 99.75% quality steam at 2.5 MPa to the primary side of the oil sands saline boiler (included in the CANDU supply scope here).

Oil sands recovered and treated water is supplied to the secondary side to this boiler at an inlet temperature of 160ºC, and exits at 80% steam quality (water-oil separation, cleanup and heat recovery for oil sands steam is considered to be external to the CANDU scope of supply) at 2.2 MPa. This results in CANDU steam condensate exiting from the CANDU side of the oil sands boiler at 177ºC.

As noted earlier, the CANDU steam is supplied at an initial pressure of 6.5 MPa. This enables the steam to be passed through a high-pressure turbine before being sent to the oil sands boiler. For the ACR unit, the nominal thermal output from the reactor in its standard configuration is ~ 1980 MW(th). Thus, after passing steam through the HP turbine and providing the appropriate steam flow to the oil sands boilers, there is additional steam flow remaining, which is then reheated and passed to a low-pressure turbine in parallel with the flow to the oil sands boiler. This enables an additional portion of electricity generation.

After passing through the low-pressure turbine, the remaining steam latent-heat energy must be rejected to a condenser, before the condensate can be returned to the CANDU steam generator. The total quantity of heat rejected to the above-ground environment via the condenser is approximately 350 MW(th). This is relatively small compared to the quantity of heat rejection in a conventional steam turbine plant dedicated to electricity production. (For example, application of a CANDU 6 unit to electricity production gives a total heat rejection to the condenser of approximately 1330 MW(th).)

Since water resources are very limited at the oil sands site, condenser-cooling water (CCW) must be re-circulated through the condenser, with CCW heat rejection via cooling towers. Alternatively, an air-cooled condenser is eventually mixed with the larger flow of hot condensate from the primary side of the oil sands boilers, thereby eliminating the need for low-pressure feedwater heaters. Using this approach, the CANDU unit operates with a water-balance with the site. The only water requirements would be modest make-up water to the CANDU boiler, within the capacity of the recycled water supply to a typical SAGD site.

It is important to note that at an early stage in project definition, the option is still available of adjusting the total output of the unit, and the steam supply temperature and pressure, to optimize the overall combination of the oil sands steam supply. Options 1 and 2 shown in Table II give alternatives based on a minor increase in CANDU unit output.

Site selection for the CANDU plant would involve a site area analysis in order to determine the location of the plant. This stage would involve an evaluation of parameters such as foundation conditions, construction material availability and transportation.

In addition to geotechnical and groundwater conditions, one critical parameter would be the determination of a minimum distance between the plant and the SAGD wells such that the bitumen extraction process would not create any hazard to the foundations or the structures.

In view of the expected geotechnical conditions at the proposed site, it is proposed that the major nuclear island structures (reactor building and reactor auxiliary building) be supported on a common foundation raft. The common foundation raft concept will alleviate any potential problems of differential settlements or deformations, noting that CANDU plants are designed to resist the effects of earthquakes.

Based on this preliminary assessment, it is believed that a CANDU plant can be sited, designed and constructed at the SAGD facility area with minimum design modifications. The constructability approach of the plant would be similar to the practice in construction of other (both existing and planned) oil sands facilities. For some proposed projects, the individual wells are planned to be arranged around a single central distribution source, with a radius of one or two kilometres, which would be very compatible with the ACR. In other cases, where steam distribution may be over several kilometres, the ACR configuration (e.g. choice of steam pressure) would have to be carefully chosen, to minimize energy losses and distribution costs.

7. CO-GENERATION STREAMS

7.1. Oxygen production

If electrolysis were deployed for H_2 production, oxygen will also be produced. The only local application identified for its use is to enrich the air feed to Claus desulphurizers. This is an established approach [8] to increasing the throughput of Claus units by between 30 and 55% (depending on H_2S concentration) from a doubling of O_2 enrichment above natural. The capital cost of Claus units is around 200 M$ and so adding O_2 to their feed would appear to have significant potential to save capital. However, the beneficial effect is considered too hypothetical and *zero* value has been assigned to the oxygen for the base case. The effect of saving half the cost of a Claus plant has been examined in a variant financial case. Note, however, that intermittent operation of electrolysis would require oxygen storage.

7.2. Heavy water production

At around 100 MW or more of electrolysis, the co-production of heavy water (D_2O) by the CECE process is expected to enhance the economics of electrolysis. A CECE unit has been included in the analysis of the economics of all cases. It should be noted that D_2O could also be co-produced with an SMR using AECL's Combined Industrial Reforming and Catalytic Exchange (CIRCE) process.

8. ECONOMIC ANALYSIS

The reference set of economic assumptions was used for project evaluation, and the evaluation of CANDU nuclear energy supply has been made as follows:

— Establish a reference target price for steam supply to the oil sands production facility (chosen as a percentage – less than 100% – of the price derived from a reference natural gas price).
— Based on a preliminary conceptual estimate of the CANDU energy supply system capital and operating costs, calculate the Internal Rate of Return (IRR) for the total CANDU capital investment, for comparison to typical threshold acceptance rates used in project development.

Calculate adjusted values of IRR for two alternative scenarios to compare to the reference economic scenario above for both optimistic and worst-case economic scenarios.

This three-scenario assessment has been performed for the various CANDU energy supply configurations.

An appropriate approach to evaluating the alternatives, to account for both capital and operating cost differences, is to choose the appropriate IRR for the project and to determine the cost of electricity that would be needed to yield that return.

The worst-case scenario results represent an estimate of minimum credible steam and electricity prices over a sustained multi-year period given the evaluation framework where all dollar values are expressed in year 2000 dollars, and uniformly escalated – i.e. no relative changes in price with time are considered). The results show that, even for this scenario, representing an adverse economic outcome, a CANDU project would still earn a significant overall return on investment.

In contrast, the Optimistic scenario represents the potential for real future price increases for natural gas and electricity. For example, the Optimistic scenario electricity price is about equal to the average Alberta pool price for electricity this year. In practice, the Optimistic scenario does not represent a ceiling on electricity or natural gas prices; rather it represents the additional return on investment, which a CANDU project might earn under a range of economic outcomes that maintain the current levels of demand for electricity and natural gas. Higher prices still can be considered under many economic scenarios. The scenario also represents an approximate value of natural gas price required to sustain investments in new natural gas sources such as gas pipelines from arctic production.

Thus, the three scenarios are chosen in such a way that actual returns on investment would be expected to achieve values between the Reference and Optimistic scenarios. The existence of positive IRR values for the Worst Case scenario represents assurance that, provided the CANDU energy supply is constructed and performs according to plan, the investment will continue to be profitable under all price circumstances.

Most recently, AECL commissioned an independent study by CERI (Canadian Energy Research Institute) to compare the economics of ACR-supplied energy with natural gas. This study provides a more up-to-date evaluation using most recent assumptions on configurations and economics and compares identical energy supplies from the nuclear and natural gas options respectively to ensure an apples-to-apples comparison. The study identified comparable ACR and natural gas supplied configurations each delivering both steam and electricity. This enabled the optimum gas configuration cogenerating electricity and steam for a typical large-scale SAGD operation. A common economic model was also developed, using parameters such as gas and electricity costs based on recent norms, but without attempting to extrapolate or forecast future prices.

Table II. Calculated internal rates of return for CANDU energy supply system

	ECONOMIC SCENARIO		
CANDU Configuration	Optimistic	Reference Case	Worst Case
Option 1	20.1%	15.4%	12.7%
Option 2	22.1%	17.2%	14.4%

The results of the CERI study [9] (which are discussed by Hopwood *et al. [10]*) show that, based on a gas price of Can$4.25/GJ (equivalent to US$3.25/million Btu reference price on the NYMEX commodity exchange), the nuclear option achieves a 10% advantage in steam cost. For comparison, current 2003 gas prices have averaged US$5.76/million Btu so far, and long-term prices are predicted to trend in this direction based on the costs of LNG imports and arctic gas supply. The study also looked at energy price sensitivity to changes in key parameters and the potential nuclear advantage is evident in a lower sensitivity to main parameters. For example, a 25% increase in capital cost (the most significant nuclear parameter) would increase steam costs by 20% from $8.61/tonne to $10.30/tonne, while a 25% increase in the price of natural gas would increase steam cost from $9.42/tonne to $11.78/tonne. In practice, the volatility of natural gas prices has been significantly greater than this. The result shows that, in this regard, nuclear has a significant advantage in cost risk.

It should be noted that the impact of CO_2 of emissions costs has been assessed, using a nominal reference value for carbon dioxide credits. The current reference basis for CO_2 costs or credits is the value of $15/tonne CO_2, stated by the Canadian government as an envelope to short-term CO_2 costs in its announced Kyoto strategy. This CO_2 cost would add 18% to the cost of natural gas-supplied steam.

Setting this carbon credit value to zero reduces IRRs by up to 1%. However, potential values of credits, forecast to support compliance with Kyoto targets, could be several times this reference, so that application of increased values of carbon dioxide credits strongly increases IRR.

The results of the CERI study state in part: "Steam supply from an ACR 700 nuclear facility is economically competitive with steam supply from a gas-fired facility. Steam supply costs from a gas-fired facility are very sensitive to natural gas price and Kyoto compliance costs.

9. DISCUSSION RELATED TO FUTURE NUCLEAR POTENTIAL, HYDROGEN AND COGENERATION

The above analysis shows that electricity prices of 20 – 30/MWh are required for electrolysis to be competitive with SMR production of H_2. Much higher prices for natural gas or large credits for CO_2 – emission avoidance would be required to change that situation. There are two possible circumstances that could make electrolytic H_2 much more competitive.

— Attributing value to the O_2 (as capital avoidance) has strong leverage. As already noted, if the quantities of O_2 needed to augment the performance of Claus desulphurizers is relatively small compared to the quantities of H_2 considered in this study, electrolytic H_2 could be much more attractive.
— The other circumstance that could change the position of electrolytic H_2 would be a substantially lower electricity cost. Recent volatility in the pool price of electricity in Alberta makes any quantitative assessment of this very uncertain. The capital cost of additional cells is not a large impediment but storage costs could be. With the fairly superficial information available, avoiding periods of peak power costs does not appear greatly to enhance the economics of electrolytic H_2. A combination of SMR and electrolysis to enable the latter to avoid periods of high-priced electricity is an alternative with some promise.

Favouring SMR-produced H_2, would be:

— Higher values for electricity; and
— Assigning the CO_2 avoidance credit to the CANDU's generation of the electricity.

For future study, the additional value attributed to the co-produced O_2 to augment Claus plant desulphurizing should be assessed, as this would improve electrolysis competitiveness. It is worth noting that, should electrolytic H_2 become a committed market for CANDU-generated electricity, this would largely eliminate future price uncertainty for the H_2 supply.

Reference [1] estimates that, using a conservative screening process, 28 oil sands projects can be identified in the northern Alberta/Athabasca region. The total energy required for such a scope of oil sands extraction has been estimated as about 60% of the conventional reserves of natural gas in Alberta. Clearly there is important value in developing alternate sources of steam supply. The nuclear oil sands proposal opens up future opportunities:

— At the end of the oil sands project life, the nuclear unit would have significant economic life remaining. After plant refurbishment, the CANDU could continue operation for a further 30 years to supply a new oil sands production or to be adapted as an electricity generator.
— CANDU technology can be readily applied, via additional CANDU units, to additional oil sands extraction projects. Participation in a successful first application offers advantages for stakeholders, via readiness to exploit CANDU technology in future oil sands technology.

The scale of possible development of future oil sands projects is extremely large. Therefore it is important to consider energy supply alternatives with regard to future expansion of the same technology, and with regard to principles of sustainable development. This is consistent with Canada's commitment to sustainable development and to the Kyoto Protocol.

The Nuclear energy supply option is highly compatible with the objectives of sustainable development, with productive use of resources to meet today's needs without compromising the needs of future generations. A CANDU reactor uses less than 100 tonnes of uranium/year to generate the energy for a typical project. Canada's estimated reserves of uranium (1997 at <$130/kg), centred on the Athabasca basin in Saskatchewan, are about 430,000 tonnes. This is large enough to supply Canada's current CANDU nuclear plant needs for hundreds of years. In addition, known global reserves at the same cost are about ten times as large. The ultimate recoverable total global supply of uranium at higher prices is virtually inexhaustible. The average abundance of uranium in the earth's crust is 2.7 parts per million (more common than tin) and it is widely available in ore bodies at various concentrations. Because fuel cost is such a small fraction of the energy cost from CANDU, nuclear energy generation is economical at high enough prices to assure long-term supply.

The nuclear option protects the environment and landmass, since the nuclear plant land requirements are small, and construction and operation will have an impact only on a limited local environment (no pipelines or major rail transportation corridors are required for fuel supply). Since the uranium requirements are small, the land area for mines to support the plant will be small.

During the operation, the CANDU plant emits no harmful sulphur or nitrogen oxides, and no CO_2 or other climate-change gases. The full life-cycle production of CO_2 for a CANDU plant (including energy used during plant construction) is less than 1% of that from a comparable gas-fired power plant averaged over the plant lifetime. Because of the enormous scale of future oil sands production, it is crucial to establish greenhouse-gas-free extraction methods at an early stage. CANDU plants meet this need.

We can ensure a positive legacy for the future by also including the development of Generation IV nuclear systems that can cogenerate electricity, process heat and hydrogen (as well as desalinated water) for an oil sands application.

10. FUTURE NUCLEAR ENERGY EVOLUTIONS

Future reactor technology (and hence CANDU technology) is expected to continue to evolve to produce lower capital-cost, higher efficiency (higher steam temperature) designs, so-called Generation IV systems [11]. Significant changes in CANDU steam supply characteristics are expected to occur over a period of two decades so will be applicable to later oil sands development.

One such possible system evolution is the CANDU variant of the SCWR (Super Critical Water Reactor) where higher temperatures and pressures are achieved. Our analysis of the costs of hydrogen from such future systems (assuming a capital cost of < \$1000/kW) is shown in Figure 3. While the thermochemical S&I process (which dissociates steam into hydrogen and oxygen using dissociation of sulphuric acid and hydrogen iodide at temperatures up to 800°C) looks to be an unpromising application for high-temperature nuclear heat, using such heat to replace burning of natural gas for an SMR has considerable scope to reduce the sensitivity of SMR-produced H_2 to the cost of natural gas.

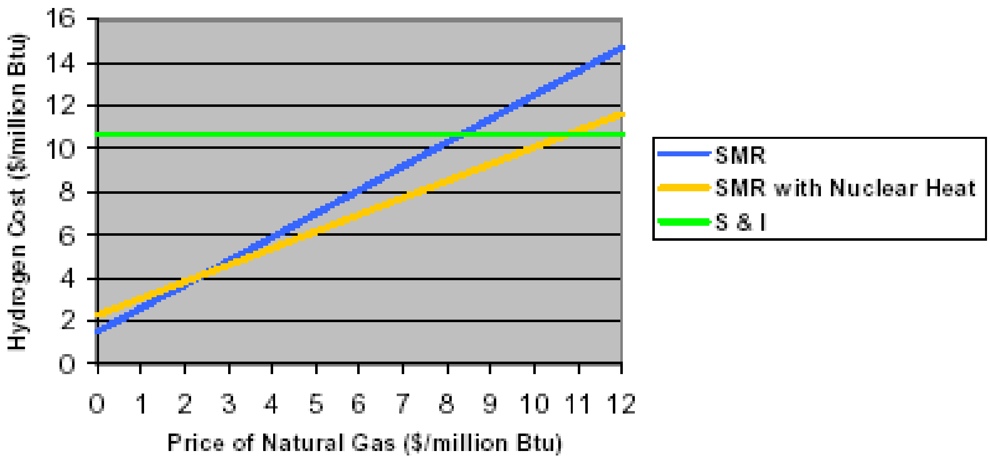

FIG. 3. Effect of natural gas price on cost of H_2 production by various processes.

11. CONCLUSIONS AND RECOMMENDATIONS

Based on a transparent reference set of economic assumptions, a new ACR nuclear steam supply plant could deliver steam to oil sands production facility at a substantial price reduction compared to the reference alternative (natural gas), while still earning an overall Internal Rate of Return of 15%.

The nuclear energy system such as an ACR could be brought into commercial operation before the end of this decade, at a date that would enable many phases of SAGD oil sands extraction to be supplied cost-effectively. The CANDU energy system is practical to implement at an oil-sands site, and is readily adapted for co-generation of steam and electricity.

While a CANDU would co-generate around 250 MW of electricity as well as the primary steam product, the displacement of SMR-produced H_2 by electrolytic H_2 has its own stand-alone economic case. Using new high efficiency technology, the economics of electrolytic H_2 could surpass those of SMR H_2 using plausible, though favourable, combinations of the main factors of (a) natural gas cost and (b) value for co-product O_2. Electrolytic H_2 likely warrants further evaluation separate from steam supply by a CANDU.

The results in this paper indicate that an ACR may be a highly competitive energy supply option for oil sands development within the next decade. This nuclear option is suitable for detailed studies covering engineering; component development and testing; licensing; and environmental assessment.

In addition, to this near-term potential, the future developments in Generation IV and CANDU reactor technology suggest that there is further potential beyond 2020, which could ultimately lead to the use of advanced co-generation concepts and systems.

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CORE DESIGN OF INDONESIAN EXPERIMENTAL POWER REACTOR

AS NATIO LASMAN, SOEDYARTOMO SOENTONO, ABU KHALID RIVAI

BATAN
Center for Development of Advanced Reactor System
NNEAI, Jakarta, Indonesia

Abstract. The core of the Indonesian Experimental Power Reactor (IEPR) has been designed with the power output of 330 MWth or 100 MWe. This design was based on neutronic behavior especially in two main analyses, i.e. firstly, utilization of fuel with *coated particles* (CFP, *Coated Fuel Particles*) 4,8% enriched and secondly, fuel with non CFP 5% enriched. The CFP is filled into the Cirene type fuel pin and arranged into a bundle of 17 x 17 array. The CFP and non CFP fuels are placed into a core with reference of SMART reactor. Various H2O moderator condition was reviewed and some core physics parameters were analyzed, i.e. the multiplication factor, Doppler effect and neutron flux distributions. The core multiplication factor of both fuel types to be decreased with the increasing of void fraction. The calculation shows the IEPR core reactivity was better than the non CFP core. The core temperature coefficient is negative and this inherent negative temperature coefficients are better from core safety aspect point of view. The neutron flux distribution is quite smooth meaning better sfor core fuel management.

1. INTRODUCTION

At present Indonesia does not have a power reactor, only 3 research reactors are available. These reactors have different thermal power. The ones with thermal power of 100 kW and 2 MW, both are TRIGA type reactor. The other one is 30 MW MTR type reactor. The 100 kW research reactor is located in Yogyakarta, while the 2 MW reactor is in Bandung, and the 30 MW research reactor is located in Serpong, Tangerang. All 3 places are in Jawa island. The research reactors have been used for material testing, and radioisotope production especially the ones of 2 and 30 MW.

The core design of the Indonesian experimental power reactor (IEPR) has been calculated as an R & D program utilizing on the experience in 10 MW isotope production reactor design and HTR calculations. While the isotope production reactor is designed using MTR fuel type, the IEPR is designed for utilizing coated particle [1] which will be filled in the cladding of Cirene fuel type. This activity has been done since August 2002.

The IEPR is designed to have an output of 330 MWth or about 100 MWe. The utilization of Cirene fuel cladding does not mean that the IEPR must use D2O as moderator, in this case the moderator is still light water. On the other hand the utilization of coated particle is not merely to increase the core safety, but hopefully could increase the moderation factor through the double moderator materials i.e carbon and water where the location is separated by fuel cladding.

2. CORE DESIGN

2.1. Nuclear fuel

The Cirene fuel type installation was built in the beginning of 1980 beside of MTR fuel element production installation This fuel type is aimed for heavy water reactor. Slightly enriched fuel of up to 5% could be experimentally produced in the Cirene fuel type installation. This installations are located in Serpong, Banten Province, Indonesia. Another activity is R & D in field of coated particle for HTR . This activity has been done since 1996 in Yogyakarta facilities.

The calculation has been done for UO2 coated particles being filled in the Cirene type fuel pin. The advantage of this fuel is to minimize the fission product release because more barriers are there, i.e. the TRISO coated particle (Fig.1), the carbon layer and the cladding. The disadvantage however is the necessity to lower operating temperature due to the fuel cladding limited capability. The IEPR operating core temperature will not be as high as the one of HTR, however it still be compatible to LWR and HWR core operating temperature of $300 – 400^{o}$ C.

These fuel pins are arranged in form of fuel assembly in 17x17 array. H2O is employed as moderator. This arrangement shows the result that this core corresponds to PWR type reactor than BWR [2, 3, 4]. This is concluded due to the fact that the multiplication factor is lower if there is a bubble in the moderator (negative reactivity). More bubbles in the moderator means that the core reactivity is towards subcritical.

The thickness of Cirene fuel cladding [5] is 0.8 cm. The cladding material is Zircalloy. The inner radius of the pin cladding is 0,94 cm and filled with the coated particle of 4.8% enrichment. Thin carbon layer is used on the inner surface of the cladding. The porosity between coated particles is used to accommodate the fission gas release.

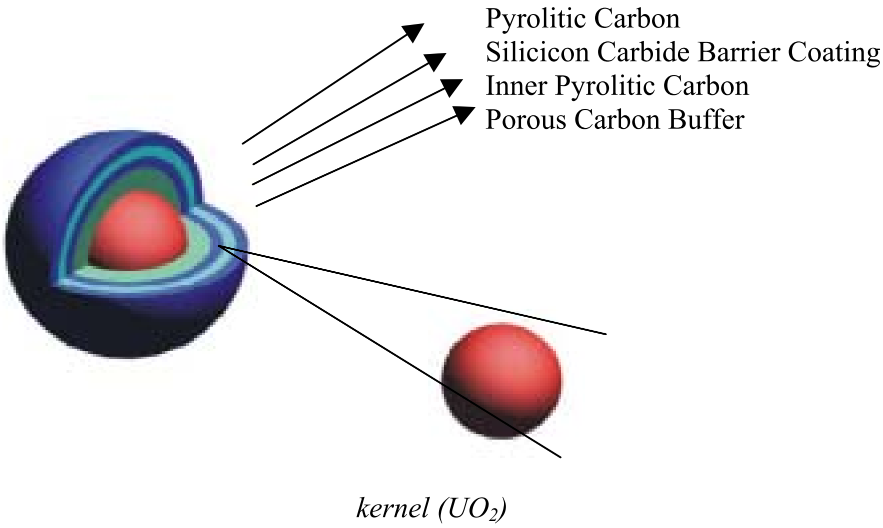

FIG. 1. Coated particle.

2.2. *Core reactor*

The reactor core must be designed at such a power level so that the heat removal system will be excellent, the temperature of the fuel and cladding anywhere in the core must not exceed the safe limits. Otherwise fuel element damage might result in releasing of radioactive material into the coolant, or even worse situation, i.e. core-fuel meltdown. The thermal limitations on reactor power have obvious effects on nuclear power economics. In the calculation, we can define this safety limits through the ppf (power peaking factor) of its fuel location, that occur due to the variations in normal or transient operation of neutron flux from a core average value. Another factor is the flow distribution factors. A decrease of coolant flow causes a decrease in the heat transfer coefficient and causes higher fuel temperatures. In this calculation the flow distribution factors will be further determined in the next step activities.

The fuel assemblies are arranged in a cylindrical form. 57 fuel assemblies are used. A total diameter of the core is 180 cm, and the active height is 200 cm similar to SMART reactor [6] (Table I). The core is designed for 330 MWth. The moderator and core cooling are the same material, i.e. H2O. The calculation was done without the influence of the absorbers.

3. CALCULATION METHOD

The calculation is done employing SRAC-95 EWS (Standard Reactor Analysis Code Engineering Work Stations)[7]. The calculation scheme is shown in Figure 2. The cell geometry is determined based on the nuclear fuel composition with 4.8% enrichment for coated fuel particle (CFP) and 5% for non CFP. By using water as moderator, the macroscopic cross section was calculated, the modell is shown in Figure 3. Fission cross section is calculated by using Collision Probability Method (PIJ) in form of 2 dimension cylindrical geometry by using CELL modul where the constant group data based on JENDL-3.2. Multiplication factor, power and flux distribution in core have been calculated by using CITATION.

The calculation has been done in various moderator condition, i.e. 100% in form of liquid, 90% liquid and 10% vapour, 80% liquid and 20% vapour, etc. The core temperature is set by 300^{o} C.

Table I. Core data

Parameters	CFP Core (IEPR) Specification	Non CFP Core Specification
Thermal power	330 MWth	330 MWth
Electrical power	100 MWe	100 MWe
Nuclear fuel	Coated UO2 Particle (CFP)	UO_2
Moderator and Coolant	H2O	Air
Cladding material	Zircalloy	Zircalloy
Fuel enrichment	4,8 %	5 %
Core active height	200 cm	200 cm
Core diameter	180 cm	180 cm

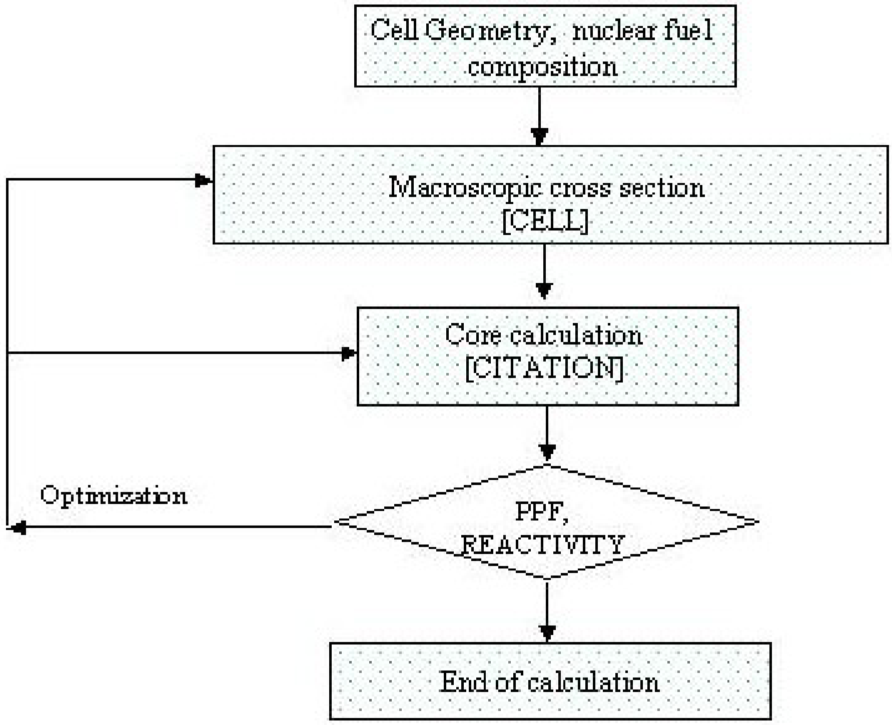

FIG. 2. Neutronic calculation flow chart.

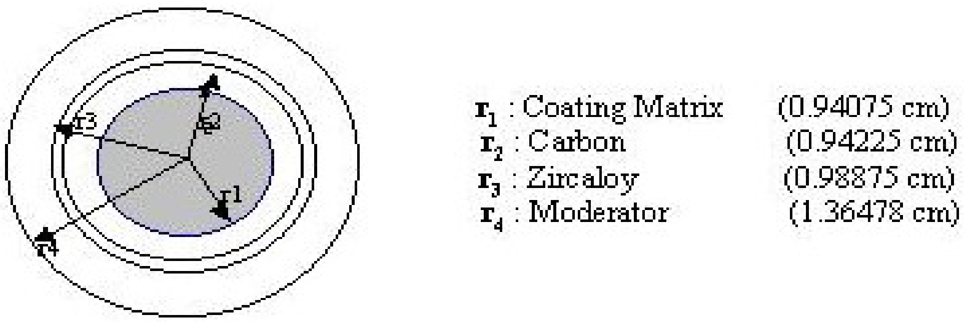

FIG. 3. Cell unit.

4. RESULTS

In this calculation, the emphasizes are on multiplications factor, Doppler effect and flux distribution. The multiplication factors (Table II) were calculated in some conditions, i.e. between liquid and vapor condition. The infinite multiplication factor for CFP is 1.31750, and for non CFP is 1.30011, it means that the moderation effect of CFP pin is better than non CFP pin. In 50% - 50% liquid – vapor condition the effective multiplication factor is < 1 in non CFP core, on the other hand the effective multiplication factor in CFP core is still positive. In the mixture of 40% liquid and 60% vapor the effective multiplication factor of CFP core has an negative value. These inherent negative temperature coefficients are better for increasing core safety aspect. It is correspond to their Doppler coefficient which is shown in Table III. The Doppler coefficient for CFP is more negative than no CFP. From this calculation it is predicted that the effective multiplication factors of CFP core (IEPR) are slightly better if the enrichment is increased up to 5%.

The axial flux distributions of both core are shown in Figure 4 and 5. The neutron flux distribution is smooth and flat enough, it means that the power distribution is smoothly spreaded and it is meaningfull for the fuel management.

Table II. Multiplication factors

No.	H_2O Moderator		k_∞		k_{eff}	
	% liquid	% vapor	CFP	Non CFP	IEPR Core	Non CFP Core
1	100	-	1,31750	1,30011	1.19515	1.19270
2	90	10	1,31212	1,27043	1.17959	1.16086
3	80	20	1,30243	1,23653	1.15832	1.12458
4	70	30	1,28700	1,19778	1.12958	1.08313
5	60	40	1,26378	1,15353	1.09095	1.03575
6	50	50	1,22962	1,10335	1.03889	0.98185
7	40	60	1,17959	1,04731	0.968252	0.92127
8	30	70	1,10541	0,98666	0.86863	0.85494
9	20	80	0,99217	0,92449	0.73417	0.78528
10	10	90	0,81100	0,86508	0.54509	0.71523

Table III. Doppler coefficient

Jenis Bahan Bakar	Koefisien Doppler
CFP	$-1,998\times10^{-2}$
Non CFP	$-9,510\times10^{-3}$

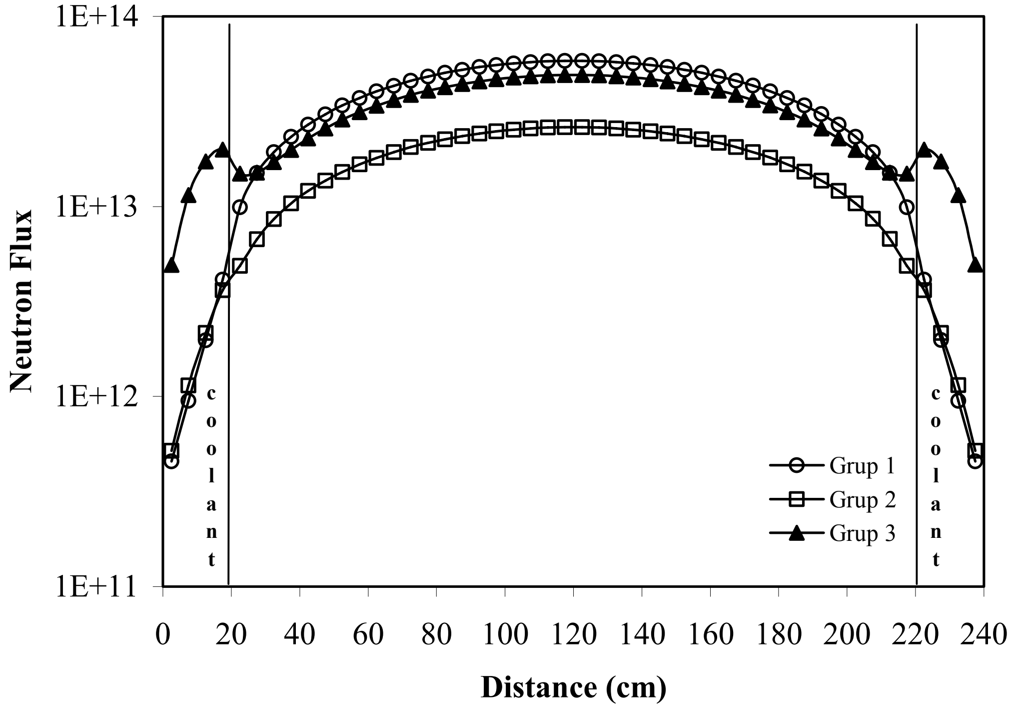

FIG. 4. Axial flux distribution of IEPR.

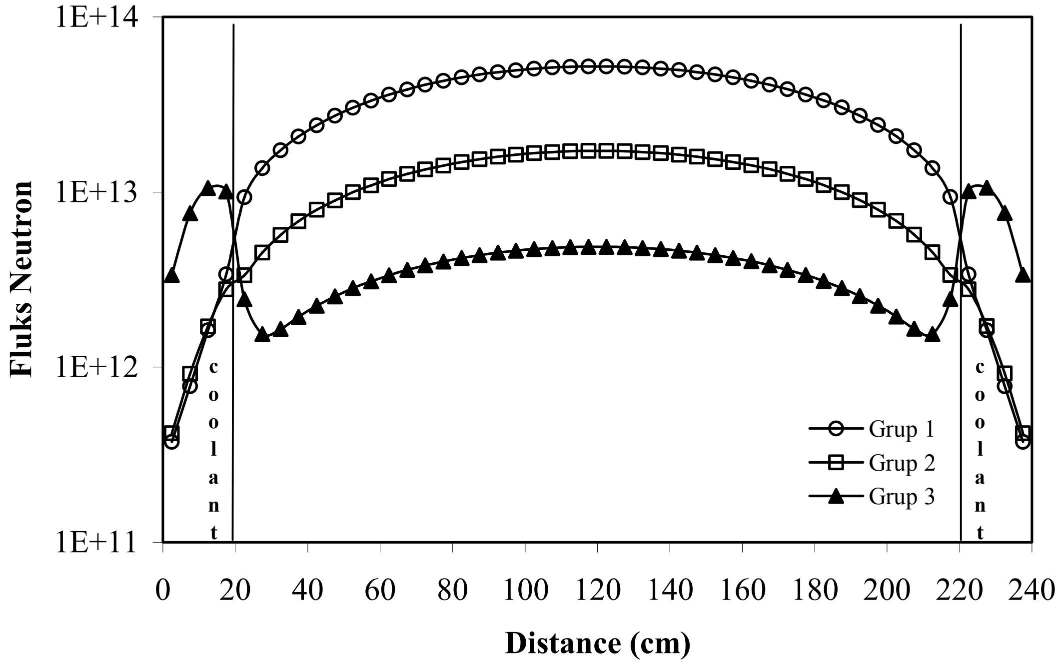

FIG. 5. Axial flux distribution of non-CFP core.

5. CONCLUSIONS

Theoretically the Indonesian Experimental Power Reactor (IEPR) core has been designed by using UO2 coated particle, which is filled in the Cirene type fuel pin. This core design is quite attractive due to the fact that:

- The barrier functions for fission product release are increased because of the performance of TRISO.
- the fuel porosity in the meat, where can accomodate the fission product release are available
- The moderation is better because the influence of carbon and water, and it causes the IEPR core reactivity is better too.
- the better inherent negative temperature coefficients means the better core safety aspect.

The next step of this R&D activitiy is the IEPR core thermohydraulics calculations except the core calculation by using the neutron absorbers.

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REDUCED MODERATION BWR WITH ADVANCED RECYCLE SYSTEM (BARS)

K. HIRAIWA[a], K. YOSHIOKA[a], Y. YAMAMOTO[a], M. AKIBA[a], M. YAMAOKA[a], J. MIMATSU[b]

[a] System Design & Engineering Department
Nuclear Engineering Center
Toshiba Corporation
Yokohama, Japan

[b] Mechanical and System Engineering Departmet
Gifu University
Gifu, Japan

Abstract. The innovative concept for a reduced-moderation spectrum boiling water reactor (BWR) combined with an advanced fuel cycle system, which is named BARS, has been being developed as one of the IVNET (Innovative and Viable Nuclear Energy Technology) development projects from fiscal year of 2000 to 2004 in Japan. The advanced recycle system means the combination of dry reprocessing and vibro-packing of MOX fuel fabrication. The reduced moderation spectrum condition in BWR is obtained through triangular tight fuel rod lattice configuration and higher void fraction. This feature cause smaller void coefficients of positive side and larger uncertainty of nuclear and thermal hydraulic design method. According to the above problems, the developing project was planned. The lattice design study by numerical analysis revealed the effect of plutonium contaminated with MA's and FP's. The two dimensional lattice design code was newly developed and applied for the evaluation of bundle nuclear performance evaluation. The location of streaming channel has been analyzed with Monte Carlo code. The evaluation showed that checkerboard location is favorable, because void coefficient becomes more negative. The critical assembly test study based on enriched uranium tight lattice showed that modified conversion ratio method is effective experimental measure for void coefficient and the current nuclear design method is applicable for tight lattice. The thermal hydraulic test has been planned and carried out in the area of single channel behavior, critical power under the high pressure utilizing small bundle and counter current flow limitation. Critical power test for tight lattice has been done utilizing the 7-rod bundle. The test result showed that existing correlation equation, Arai's equation, has insufficient applicability to predict the critical power of tight lattice. The sub channel analysis was performed. The analysis showed that flow distribution had important role in the estimation of the critical power correctly.

1. INTRODUCTION

The innovative concept for a reduced-moderation spectrum boiling water reactor (BWR) combined with an advanced fuel cycle system, which is named BARS, has been being developed as one of the IVNET (Innovative and Viable Nuclear Energy Technology) development projects from 2000 in Japan. The reduced-moderation spectrum BWR with high conversion property features higher utilization of uranium resources, multi-recycling usage of plutonium and flexibility in loading the extracted long-lived radioisotopes, it can realize a sustainable nuclear energy[1, 2, 3]. As shown in Figure1, the advanced recycle system means the combination of reduced moderation BWR, dry reprocessing and vibro-packing of MOX fuel fabrication. This system simplifies reprocessing and MOX fuel fabricating process and reduces related backend cost. The reduced moderation spectrum condition in BWR is obtained through triangular tight fuel rod lattice configuration and higher void fraction. This feature cause some technological problems; smaller void coefficients of positive side and larger uncertainty of nuclear and thermal hydraulic design method. According to the above problems, the project consists of the three parts; nuclear design study that enables the negative void coefficient, critical assembly experiment to validate nuclear design method and thermal hydraulic test to evaluate thermal hydraulic property of tight lattice.

The nuclear design study focuses on optimization of tight lattice bundle design. In the optimization of bundle design, it is important to satisfy both of the requirements for high conversion ratio and negative void coefficient. Bundle design optimization study also includes the evaluation of core performance. Streaming channel concept has been introduced to realize the negative void coefficient. Since the streaming channel concept is the important and innovative design feature for BARS core, both analytical study and experimental study on streaming effect have been planned.

The critical assembly experiment on uranium lattice has been carried out on Toshiba nuclear critical assembly (NCA) as the first step of tight lattice critical experiments. Since there are few benchmarking studies on reduced-moderation light water lattice such as BARS core, the accuracy of nuclear design method should be validated. The critical assembly experiment was planned to provide the validation data. Modified conversion ratios method have been introduced to measure the pin-wise parameter. The method is based on the gamma-ray measurement for a irradiated fuel rod.

The thermal hydraulic test focuses on three subjects related with the cooling mechanism of tight lattice. The first subject is elucidation of mechanism of the boiling transition (BT) on tight lattice. It is well known that BT is very important thermal hydraulic issue on BWR. The single channel visible experiments were planned to investigate the mechanism of the BT on tight lattice. Glass rods coated with transparent electric resistant heater material are adopted in this experiment. The two-phase (steam and water) flow behavior in the test channel has been recorded by video camera and analyzed. These experiments have been done under the atmospheric pressure in order to make it easy to observe. The second one is establishing of the BT correlation equation on tight lattice bundle. Mini tight lattice bundle experiments were planned to study the correlation. These experiment include 7-rods hexagonal bundle experiment and 14-rods square bundle experiment. Various rod configurations on tight lattice have been examined in 7-rods bundle experiments for the parametric study on tight lattice. The 14-rods bundle experiment aims the measurement of square channel effect. One of the important purposes of these experiments is to verify the existing the BT correlation equation such as Arai's equation on tight lattice bundle.

These experiments have been done in Toshiba's high temperature and high-pressure BWR experimental facility for stability and transient test (BEST). The third one is cooling performance test for tight lattice in hypothetical accident evaluation. Counter-current-flow-limitation (CCFL) experiment was planned for this purpose.

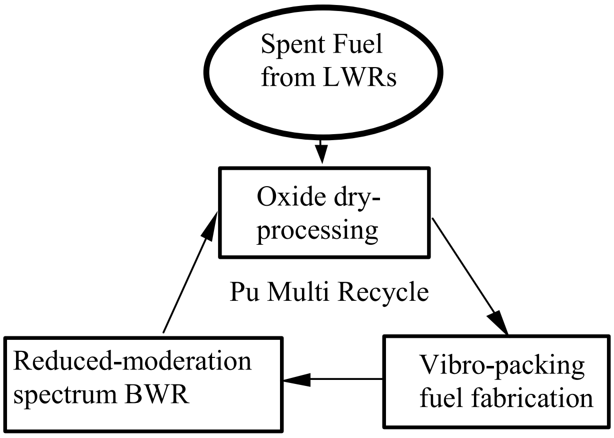


FIG. 1. BWR with advanced recycle system (BARS).

2. CORE DESIGN CONCEPT

In the design of BARS core and fuel, a tight lattice fuel assembly has been adopted where a water to fuel volume (W/F) ratio is about 0.5 to achieve a fast neutron spectrum.

It is well known that the void reactivity coefficient in LWRs has the tendency to be less negative in harder neutron energy spectrum. Then, a new core concept has been introduced in order to improve the void reactivity coefficient under the restriction of core diameter by adopting a neutron streaming channel described below.

Figure 2 shows the vertical view of the BARS core in a large BWR plant (reactor thermal output of 3926MWt, core height of 1.6 m). Partial fuel assemblies whose active fuel length is about half of the normal fuels are arranged by one-third of the whole core as shown in Figure 3.

When void fraction increases, the streaming channel located at the upper part of the partial assembly will enhance axial leakage of neutrons which have leaked out through the side of the normal assemblies and the top of the fuel bundle of the partial assemblies as shown in Figure 4. The cavity-can in the streaming channel not only provides a streaming path for the leaked neutrons from the fuel but also suppresses softening of the neutron energy spectrum by expelling water from the channel.

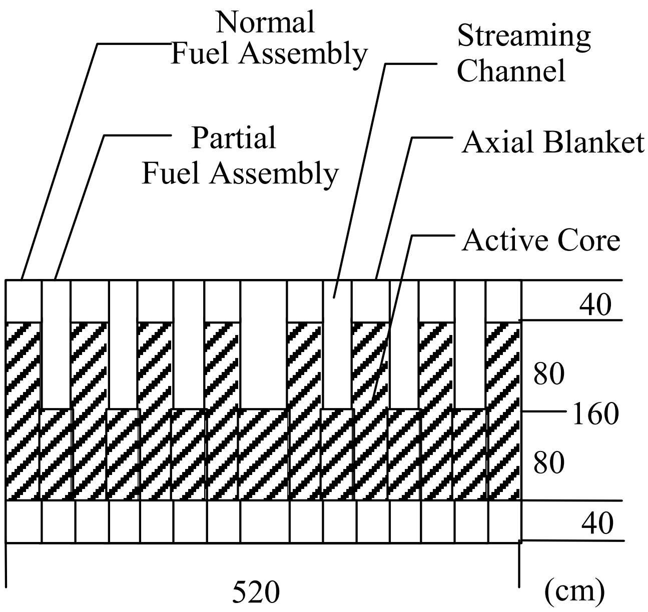


FIG. 2 . Vertical view of BARS core.

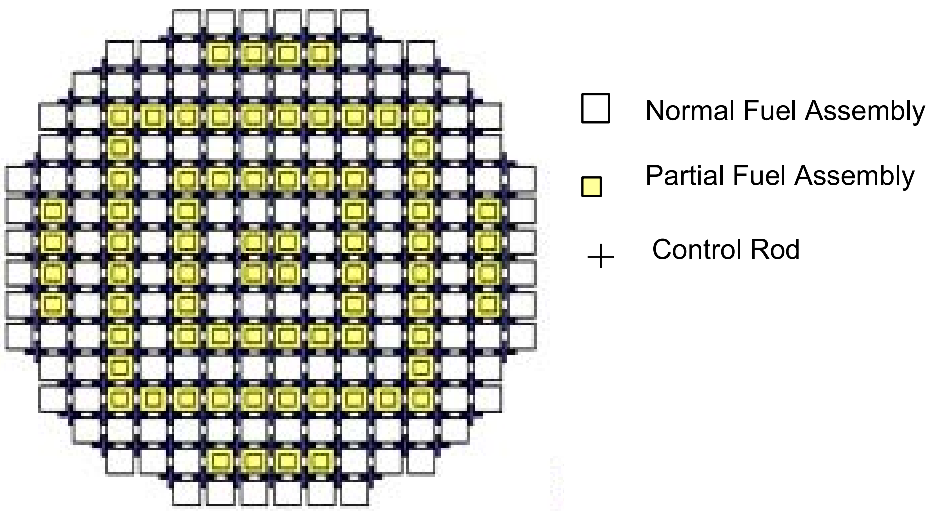


FIG. 3. BARS core layout.

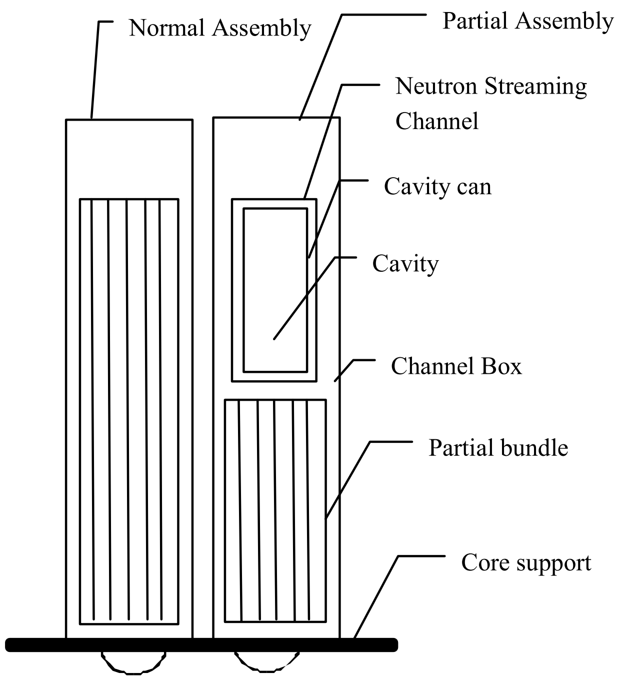


FIG. 4. Vertical view of fuel assembly.

3. NUCLEAR DESIGN STUDY

3.1. Lattice design study

The lattice design study has been done as the preparation stage for the planned three dimensional design optimization. Basic design study based on the simple pin-cell geometry and the bundle design studies based on two-dimensional geometry were performed as the lattice design study.

3.1.1. Basic design study

Since BARS core fuel assumes the plutonium extracted from dry-processing, the evaluation of the effect of dry-processing on nuclear performance is important design issue. For the purpose of the evaluation, basic design study has been carried out.

The fuel pin-cell dimension for basic design study is shown in Figure 5. The SRAC[4] code has been used for the basic design study. The conversion ratio for various hydrogen ratio based on pin-cell calculation are shown on Figure 6. Nuclear performance shown on Figure 7-10 have been evaluated under the assumption of the same hydrogen to heavy metal ratio with the bundle configuration based on Table I. In these evaluation, the reference plutonium (Pu) composition that comes from wet reprocessing is assumed. For the evaluation of the minor actinide (MA) and fission product (FP) contamination by dry-processing, decontamination factor (DF) has been assumed. The decontamination factor is defined as the ratio of the decontaminated contents to the original contents (= non-decontaminated contents). The solid line on Fig.6 is the tendency of internal conversion ratio. Although there are influences on void coefficient from various fuel parameters, it is clear that the predominant factor is hydrogen to heavy metal atomic ratio

As for the effect of contamination with MA and FP shown in Figure7, the conversion ratio decreases by about 5% when the MA whole quantity is mixed without FP contamination. When DF is greater than 70, the effect of FP contamination on conversion ratio is very small. When the DF becomes about 20, the conversion ratio decreases by 7% totally. When the DF factor becomes about 5, the reduction of the conversion ratio is about 15%.

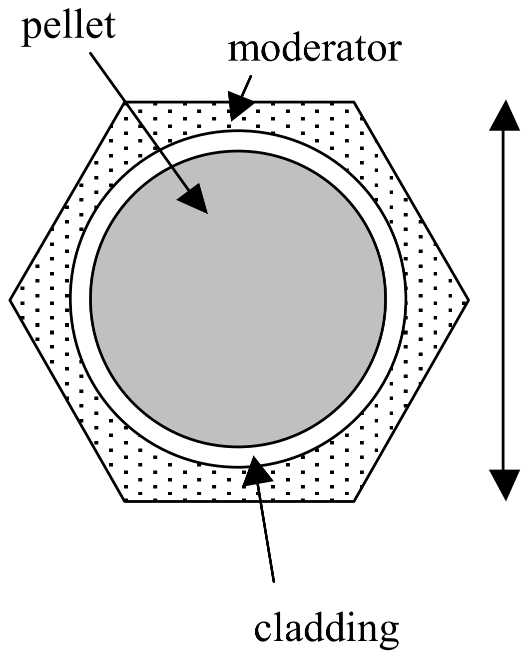


FIG. 5. Pin-cell geometry.

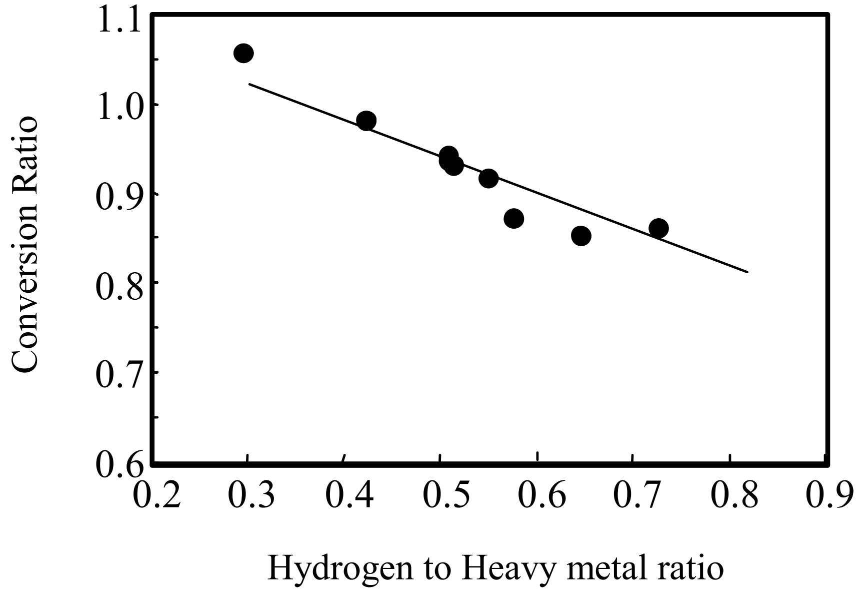


FIG. 6. Conversion ratio under the pin-cell geometry.

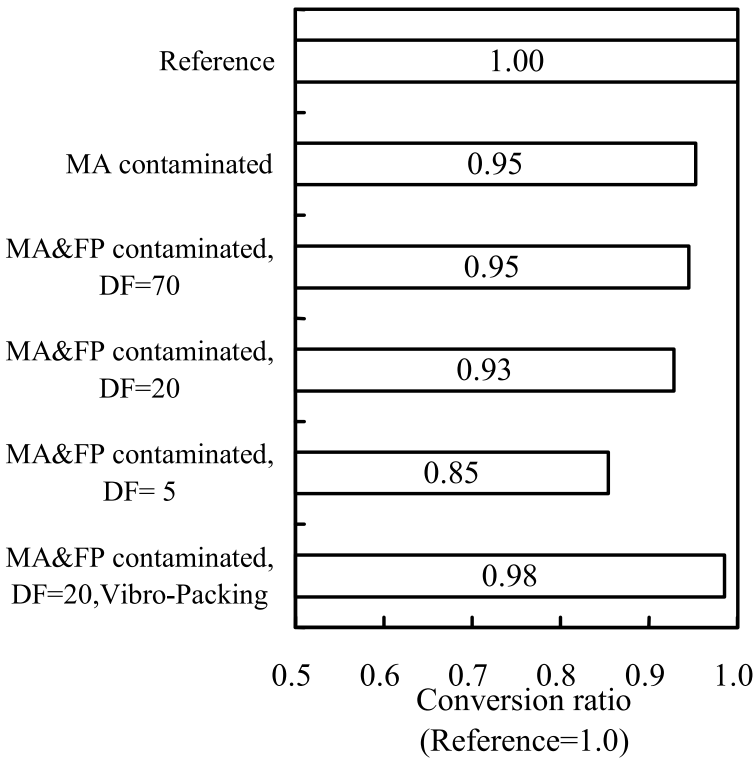


FIG. 7. Conversion ratio.

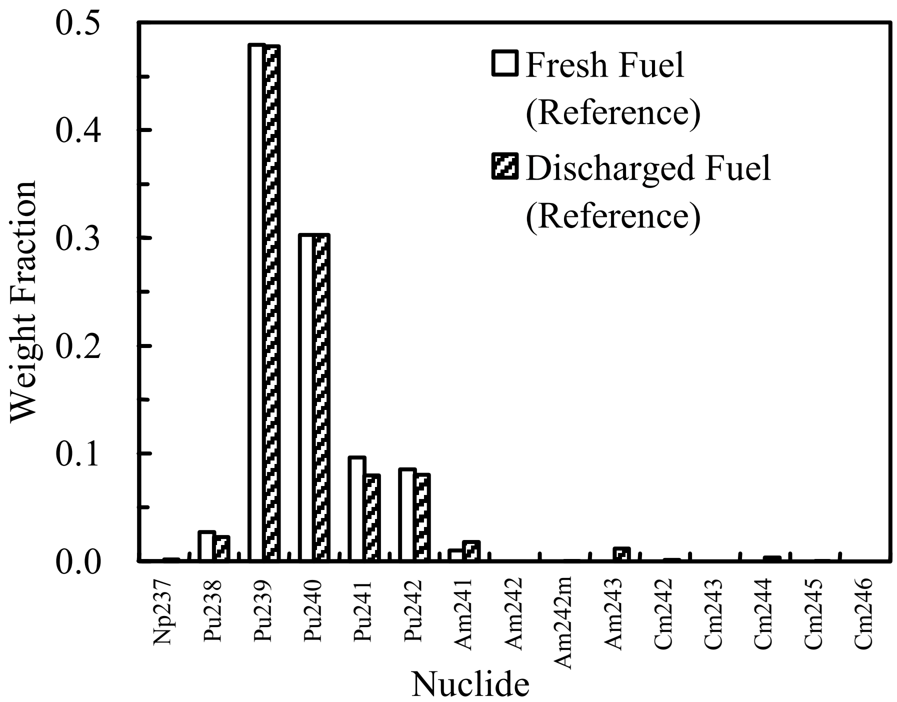


FIG. 8. Comparison of MA isotopes (Reference; No MA contaminated).

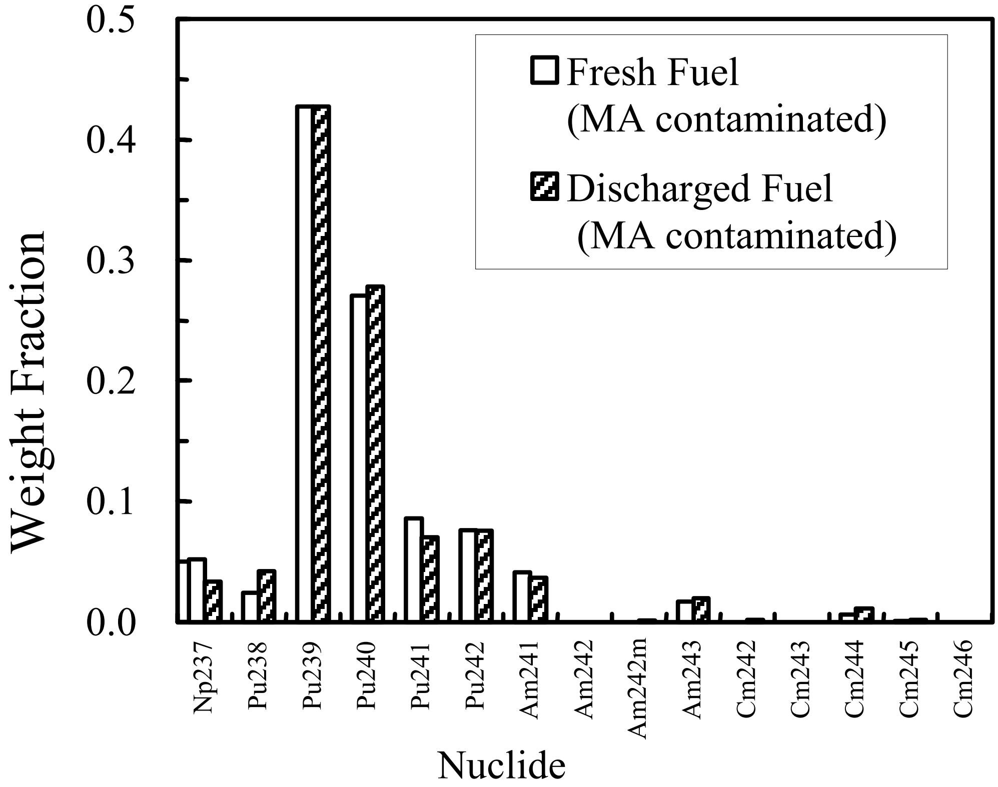


FIG. 9. Comparison of MA isotopes (MA contaminated).

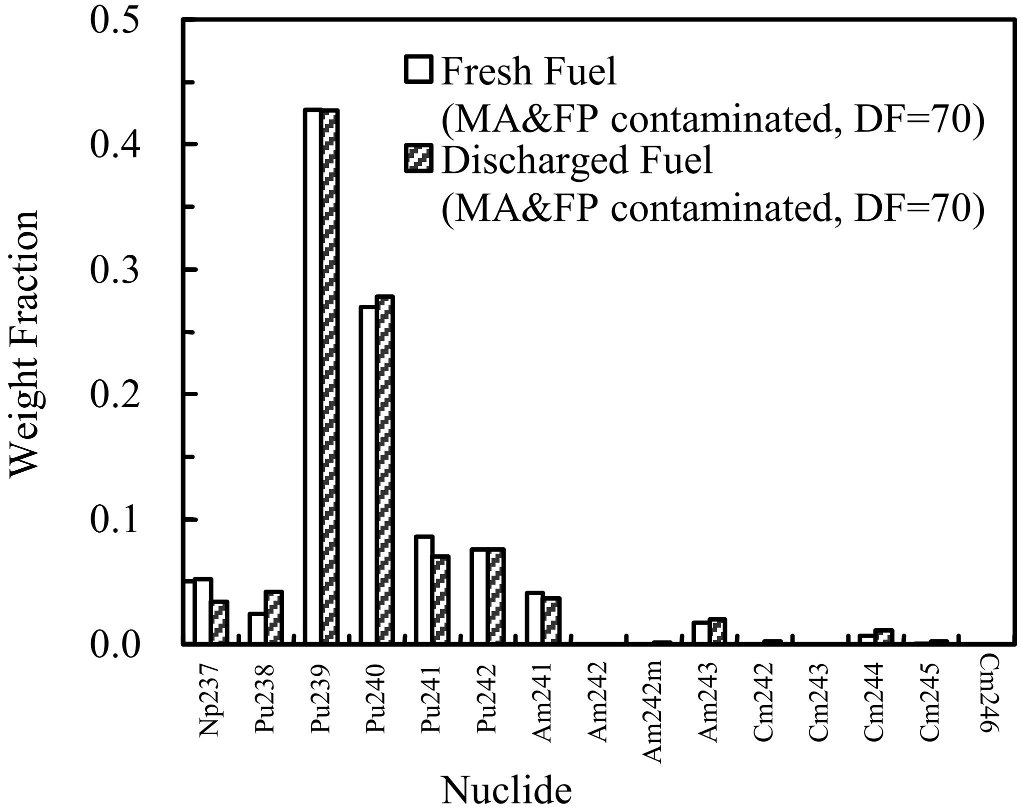


FIG. 10. Comparison of MA isotopes (MA&FP contaminated,DF=70).

Table I. Specification of BARS core

Item	Unit	BARS bundle	BWR bundle
Power	Mw	1356	1356
Core active height	m	1.6;normal, 0.8;partial	3.7
Number of assembly	-	132;normal, 76;partial	872
Number of fuel pin	-	658	74
Lattice type	-	triangular	rectangular(9x9)
Pin diameter	mm	10.8	11.2
Cladding thickness	mm	0.3(SUS)	0.71(Zr)
Pin gap	mm	1.3	3.2
Pin pitch	mm	12.1	14.4
Bundle pitch	mm	310	155
Ratio of flow area to fuel area	-	0.5	3.1

As for the effect of plutonium vactor and MAs composition shown from Fig.8 to Fig.10, vectors of initial plutonium and MAs were evaluated depend on the plutonium composition recycled from uranium fuel. The results are summarized as follows.

In reference case, contents of fissile plutonium are about 56 wt% in discharged fuel. This value is 2wt% smaller than that of initial fuel content. Similarly, the degradation of plutonium in case of MA and/or FP contamination is very small. It is clear that the influence of MA and/or FP contamination on plutonium vector is very small from above results. As for the contents of MAs, Np-237 and Am241 decrease compared with initial contents in MA and/or FP contaminated cases. Other MAs, such as Am-243 and Cm nuclides, increase slightly. From above results, BARS core is favorable for stable multiple recycling of plutonium.

3.1.2. Bundle design study

Since the lattice design calculation is required for both optimization of lattice configuration and creation of nuclear library for the three dimensional simulator, the special lattice design code for BARS bundle has been developed based on existing BWR lattice design code. The code solves the bundle geometry with triangular lattice and square channel box directly and outputs the bundle-averaged library for the simulator.

The fundamental nuclear characteristics for BARS lattice have been analyzed based on this code. Addition to the analysis of fundamental nuclear characteristics, design possibility of burnable poison and enriched uranium matrix for MOX pellet have been evaluated by the code. This study assumes the triangular rod lattice with square channel and cruciform control rod as shown in Figure 11. The detail results by burn-up calculation are shown on Figures12 and 13.

— Infinite multiplication factor in hot voided condition (Fig.12)
— Infinite multiplication factor in cold non-voided condition (Fig.13)

The analysis assumes four different enrichment of plutonium to follow the standard MOX bundle design for BWR. Under the assumption, local peaking value becomes 1.1 at the beginning of life. Although the value of local peaking is very small compared with ordinary BWR bundle, the design margin to the thermal limit related to the local peaking should be evaluated by simulator.

Burnable poison is effective measure for cold shut down performance, since the reactivity decreases at cold condition shown in Figure 3. However, the design consideration required for the adoption of burnable poison on BARS bundle because of its positive side effect on void coefficient.

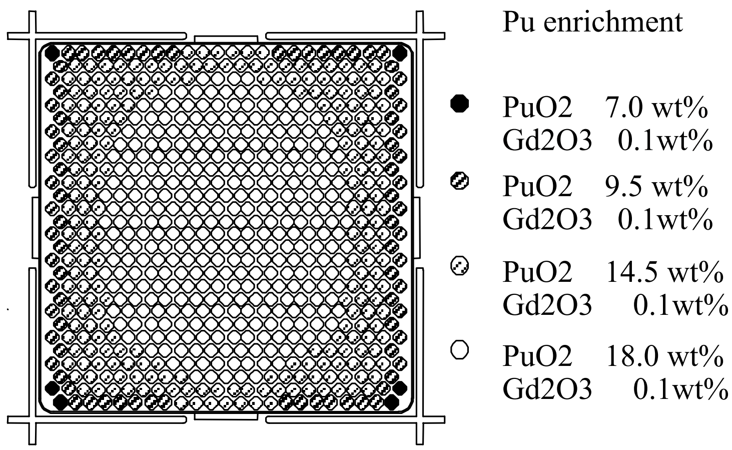


FIG. 11. Plutonium enrichment and gadolinia concentration distribution (design depleted U matrix with gadolinia.

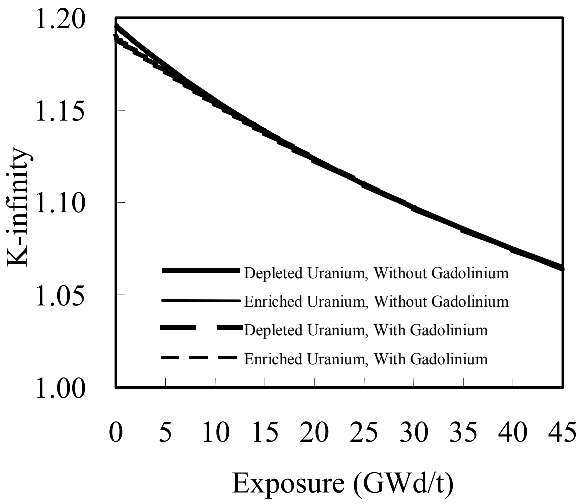


FIG.12. K-infinity (HOT).

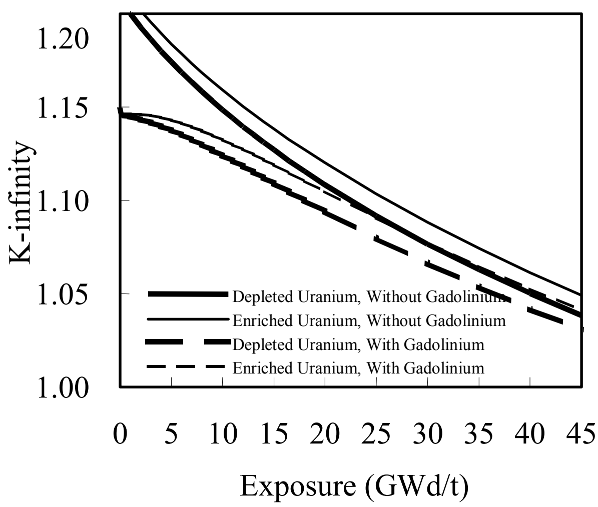


FIG.13. K-infinity (COLD).

Enriched uranium matrix has negative effect on void coefficient. Addition to this, the residual rate of total fissile almost keeps constant.

From these result, burnable poison and the enriched uranium matrix will become one design alternatives for BARS bundle.

3.2. Streaming channel design study

Since there exist various location of streaming channel, the sensitive study for the location has been done. The layered location and the checkerboard location (Fig.14) have been analyzed by three-dimensional Monte-Carlo analysis. In the analysis, typical distribution of void fraction along the vertical direction of the core and the typical exposure distribution have been assumed. Based on the assumption, the parameter study on channel gap width has been done. These results show that streaming channel enables negative coefficient and the checkerboard location is favorable compared with layered location, because void coefficient becomes more negative (Fig.15) and the conversion ratio does not become worse (Fig.16).

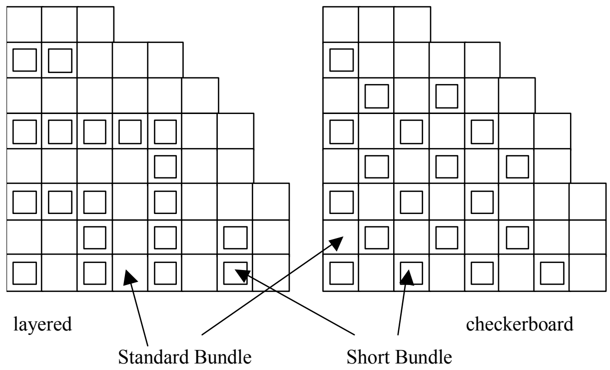


FIG.14. Loading pattern of streaming channel.

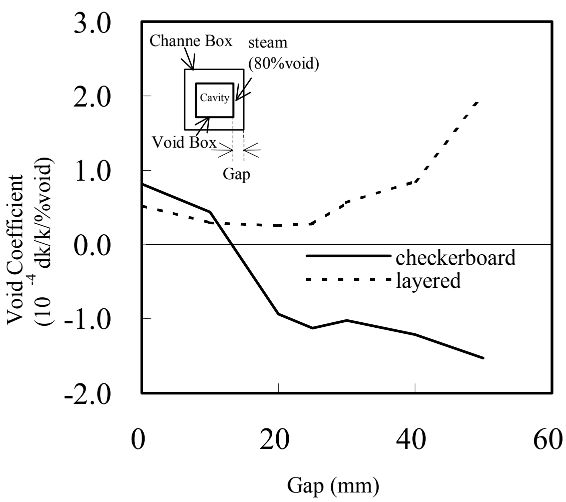


FIG. 15. Void coefficient.

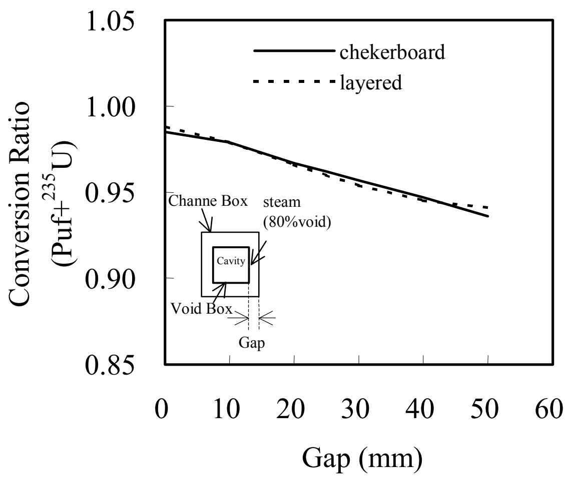


FIG.16. Void coefficient.

4. CRITICAL ASSEMBLY TEST STUDY

Several experimental studies have been made on high conversion LWR (HCLWR) with triangular tight lattice [5-6]. Chawla et al. performed experiments simulating an H_2O voidage of 42.5% and estimated k_∞ void coefficients on the basis of the measured changes with voidage of the various reaction rate ratios[5]. The BARS core has higher H_2O voidage of 60% to achieve the conversion ratio of 1.0 more than that for HCLWR. The neutron spectrum is significantly different from that in a current LWR, and therefore, applicability of a nuclear design method and nuclear data library should be validated. Doppler coefficients are also important factors for a nuclear design; however, there are a few measurements for LWR[7-8]. The Doppler coefficients for the BARS core should be validated as well as the void coefficients.

In the BARS core, resonance neutron contributes to neutron absorption and fission more than current LWRs. The neutron is also closely related to reactivity coefficients such as void coefficients and Doppler coefficients.

The BARS core features triangular tight lattices, streaming channel for keeping void coefficients negative and MOX fuels. This study focuses on reactivity coefficients for a triangular tight lattice. In fact, MOX fuels are used for the BARS core design, but this study focuses on a UO_2 core as a first step for nuclear design method validation. The validation for UO_2 fuels is also important for MOX fuels because ^{238}U capture reaction significantly affects reactivity coefficients for MOX fuels. In the next stage, MOX fuel critical experiments are planned.

4.1. Description of modified conversion ratio

Nakajima et al. defined a modified conversion ratio and measured this quantity in water-moderated low-enriched UO_2 and MOX cores[9-10]. A "modified conversion ratio" is different from a conventional conversion ratio and is defined as a ratio of ^{238}U capture rate to total fission rate. The total fission rate is due to ^{235}U and ^{238}U in a UO_2 fuel, and due to ^{235}U, ^{238}U,^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am in a MOX fuel. A modified conversion ratio can be more easily measured than a conventional conversion ratio since the total fission rate can be measured by gamma ray emitted from some fission products. We have developed a reactivity coefficient evaluation method for UO_2 and MOX cores on the basis of a “modified conversion ratio” measurement [11-14]. This method can determine void coefficients from only modified conversion ratio measurements without a number of reaction rate measurements as shown by Chowla[5]. A modified conversion ratio was deduced from gamma-ray intensities of ^{239}Np and some fission product nuclides (^{135}Xe, ^{143}Ce, ^{133}I, etc.). Neutron multiplication factor (k*) for the fuel rod was determined from the modified conversion ratio. A void coefficient of reactivity was determined from a change in k* resulting from change in void fraction.

4.1.1. Void coefficient

The infinite neutron multiplication factor (k-inf) is too difficult to measure directly in critical experiments. Instead of k-inf, we define here “finite neutron multiplication factor” for local reaction rates, k* [15]. This factor k* is the ratio between the rates of neutron production and absorption in a finite medium. A local reaction rate can be measured by gamma-ray spectrum analysis, the foil activation method, etc.

k* is defined as follows;

$$k^* = \frac{\int S(r)\,dr}{\int A(r)\,dr} \qquad (1)$$

k* is equivalent to k-inf in no neutron leakage condition. k* can be also defined for each fuel rod, that is, k*(r)=S(r)/A(r).

In a UO_2 fuel cell, S(r) and A(r) are shown as follows:

$$S(r) = (\nu\Sigma_f(r)\phi(r))_{25} + (\nu\Sigma_f(r)\phi(r))_{28}$$

$$= \nu_{25}(\Sigma_f(r)\phi(r))_{25}\{1+(\nu_{28}/\nu_{25})\delta_{28}\} \qquad (2)$$

$$A(r) = (\Sigma_f(r)\phi(r))_{25}(1+\alpha_{25})+(\Sigma_f(r)\phi(r))_{28}$$

$$+(\Sigma_c(r)\phi(r))_{28}+(\Sigma_a(r)\phi(r))_{NF} \qquad (3)$$

$$\delta_{28} = (\Sigma_f(r)\phi(r))_{28}/(\Sigma_f(r)\phi(r))_{25}$$

$$\alpha_{25} = (\Sigma_c(r)\phi(r))_{25}/(\Sigma_f(r)\phi(r))_{25}$$

Modified conversion ratio (C) is defined as follows:

$$C = (^{238}\text{U capture reaction rate}) / (\text{total fission rate})$$

$$= (\Sigma_c(r)\phi(r))_{28}/\{(\Sigma_f(r)\phi(r))_{25}+(\Sigma_f(r)\phi(r))_{28}\} \qquad (4)$$

k*(r) can be written as follows by employing C:

$$k^*(r) = S(r)/A(r)$$

$$= \nu_{25}D_{28}/\{1+C+R/(1+\delta_{28})\} \qquad (5)$$

$$R = \alpha_{25} + ((\Sigma_a(r)\phi(r))_{NF}/(\Sigma_f(r)\phi(r))_{25}$$

$$D_{28} = 1 + \{(\nu_{28}-\nu_{25})/\nu_{25}\}\{\delta_{28}/(1+\delta_{28})\}$$

Void coefficient can be defined as Dr/Dvoid.

Where,

$$\Delta\rho = 1/k^*(r)(\text{with void}) - 1/k^*(r)(\text{without void}) \qquad (6)$$

Dvoid is void fraction change.

Modified conversion ratio(C) and reaction rate ratio(δ_{28},etc.) can be defined for each fuel rod. As a result, k* is almost independent of leakage effect by a core buckling and fuel enrichment distributions.

Calculation has been done to clarify the dependence of C on void fraction. This model is based on the rod pitch and the fuel size of NCA. The dependence of C and other factors on void fraction have been calculated based on the model.

Figure 17 shows calculation results for dependence of each factor on water density. The figure reveals that C is highly dependent on void fraction compared with any other factor, and therefore, a void coefficient can be estimated by a C measurement. C is determined by FP and actinide gamma-ray measurements from outside a fuel rod.

4.1.2. Doppler coefficient

In the same manner as in the deduction of void coefficient, k* is defined as follows:

$$k^* = S/A = \nu_{25}D_{28}/\{1+C+R'/(1+\delta_{28})\} \qquad (7)$$

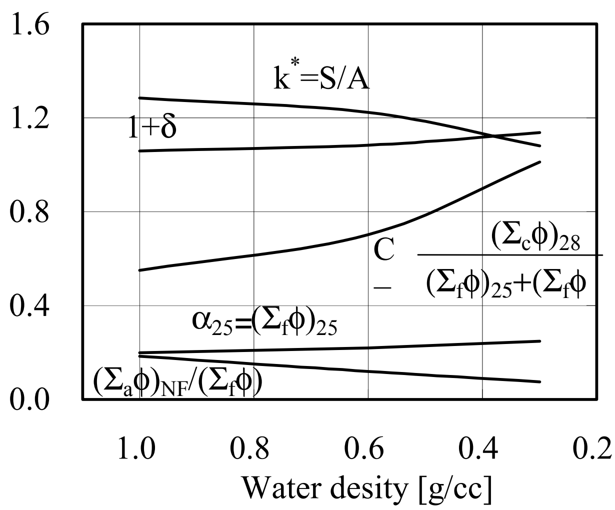
$$R' = \alpha_{25}+((\Sigma_a(r)\phi(r))_{OX}/(\Sigma_f(r)\phi(r))_{25} \qquad \text{OX: Oxygen}$$

Ratio of oxygen absorption rate to to ^{235}U fission rate, $((\Sigma_a(r)\phi(r))_{OX}/(\Sigma_f(r)\phi(r))_{25}$, is much smaller (~0.4%) than α_{25}, and almost independent of fuel temperature. The second term of D_{28}, $\{(\nu_{28}-\nu_{25})/\nu_{25}\}\{\delta_{28}/(1+\delta_{28})\}$, is much smaller (~1%) than the first term, 1, and slightly dependent (~0.03%) on fuel temperature.

Consequently, reactivity change due to change in fuel temperature (T→T'), $\Delta\rho$ (=1/k*(T))-1/k*(T')), is written as

$$\Delta\rho=(1/\nu_{25}D_{28})[\Delta C+\Delta\{\alpha_{25}/(1+\delta_{28})\}] \qquad (8)$$

Doppler coefficient can be defined as $\Delta\rho/\Delta T$



4.1.3. Gamma-ray spectrum

The modified conversion ratio is determined through a gamma-ray spectrum analysis for an irradiated fuel rod. Capture rate for ^{238}U and total fission rate are determined from ^{239}Np gamma-ray intensity and some fission product gamma-ray intensities, respectively. An HP-Ge detector has been used for gamma-ray measurements of irradiated fuels.

Figure 18 shows an example of measured gamma ray spectrum. In this study, ^{135}Xe, ^{143}Ce, and ^{133}I have been used in order to determine total fission rate.

4.2. Void coefficient measurement

NCA is a slightly enriched, uranium-fueled, light-water-moderated critical assembly, which has been utilized to validate LWR nuclear design and to develop a new fuel design concept.

Figure 19 shows the core configuration for this study. The test assembly consisted of a void-simulated zone and a driver zone. The void simulated zone is a triangular tight lattice consisting of 3.9wt% and 4.9 wt% UO2 fuels with a pile of polystyrene plates for simulating void fraction. The rod pitch of the void simulated zone is 13.5mm. The polystyrene plates have been employed to realize hydrogen concentration corresponding to 0%, 35% and 60% void fraction of a typical BWR operating condition. The driver zone surrounding the void-simulated zone used to keep criticality shapes a square lattice consisting of 2wt% UO_2 fuels with cold water. The rod pitch of the driver zone is 15.2mm. The volume ratio of fuel to moderator for cold water, hot 0% void, hot 35% void and hot 60% void are 0.59, 0.44, 0.28 and 0.18, respectively. For these four cases of void fraction, modified conversion ratios of fuel rods in the center at the void simulated zone have been measured. Gamma-ray spectra from the fuel rods have been measured with an HP-Ge detector after irradiation of 20 Watts x 30 minutes.

The measurement error in modified conversion ratio has been estimated to be 2~3%. The main components of the error corespond to the uncertainty in gamma-ray spectrum analysis, the error in gamma-ray self shielding of a pellet and the statistical error.

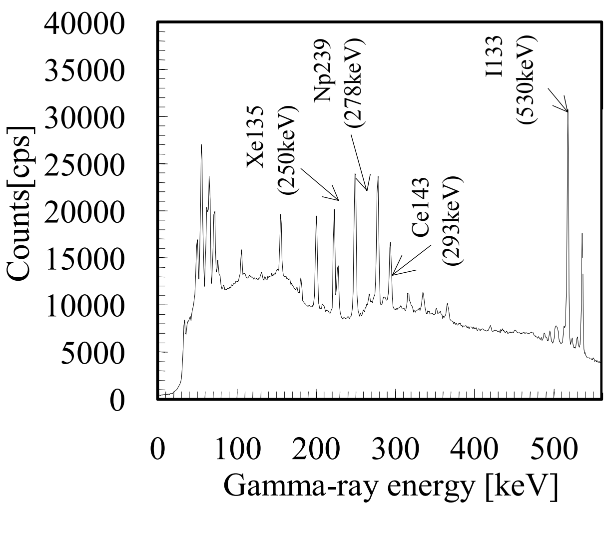


FIG. 18. Measured gamma-ray spectrum.

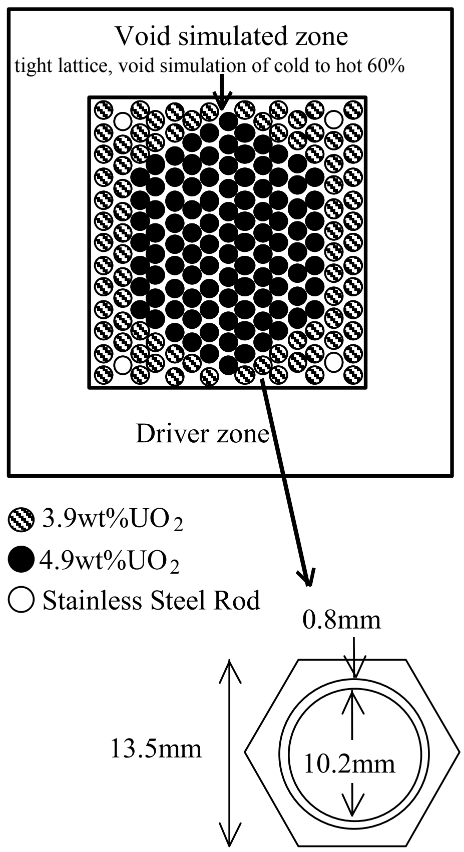


FIG. 19. Experimental core configuration for critical assembly test.

Figure 20 compares measured modified conversion ratios with calculations. Calculations were made by a Monte Carlo code MCNP4A with JENDL3.2 nuclear library [17]. It is concluded that measurements agreed with calculations within 2~3% of the measurement error.

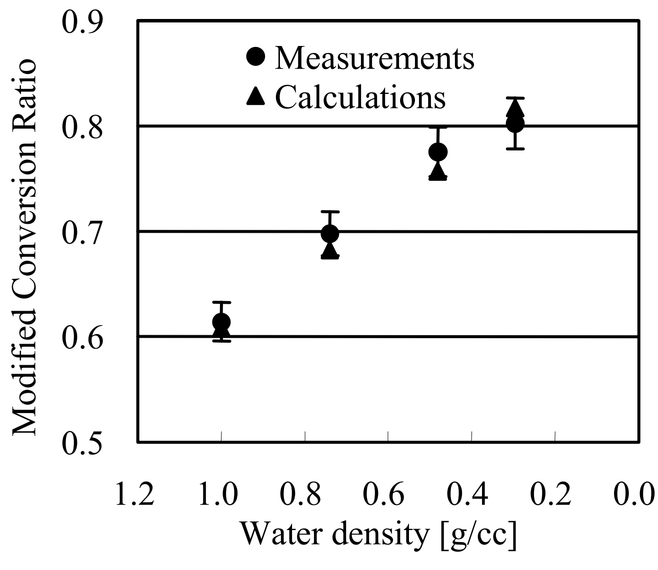


FIG. 20. Modified conversion ratio.

Figure 21 shows k*s derived from measured conversion ratios. The error of k* has been estimated to be ~1%. Void coefficients derived from the difference of k* in cold condition and hot 60% void condition was 0.153± 0.014%dk/k/%void. On the other hand, calculated void coefficient was 0.167%dk/k/%void and agreed with the measured one within the measurement error.

Consequently, current nuclear design method, MCNP4A with JENDL3.2, is applicable to BARS core design for UO_2 fuel.

4.3. Doppler coefficient measurement

The test equipment for Doppler measurement was fabricated. For Doppler measurement, it is important to raise the temperature of fuel pellets as high as possible. Moreover, the size of equipment must be suitable for the core size in NCA.

The size of the test equipment is f15cm x H 150cm, and the equipment is designed to load into the NCA core. A micro heater consisting of a nichrome wire with a stainless steel sheath is used to raise pellet temperature. Fine flex fiber composed of silica-alumina fiber is used as a heat insulator. The equipment has been designed to raise pellet temperature up to 900°K. The temperature-raising test revealed that the pellets reached the temperature of 900°K without any trouble.

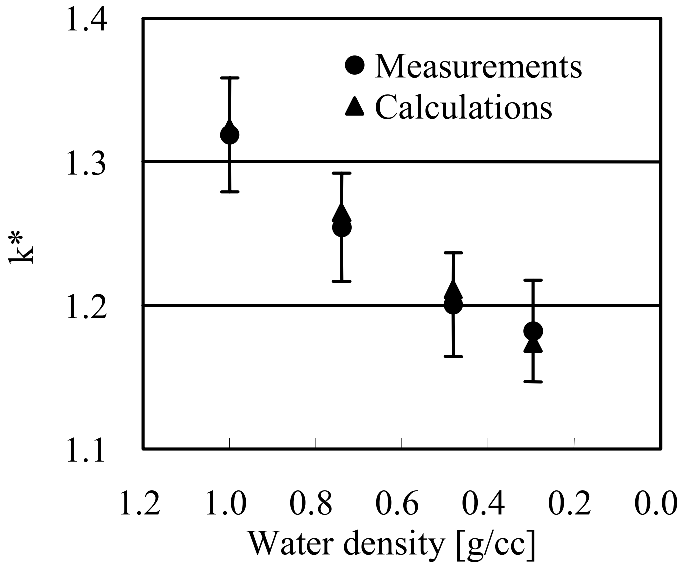


FIG. 21. Finite multiplication factor k^*.

5. THERMAL HYDRAULICS TEST STUDY

In the BARS core, a tight lattice fuel design is required because the conversion ratio needs to be improved to about 1. However, as a fuel rod gap becomes narrower, thermal-hydraulic performance, especially critical power, becomes worse. Therefore, the thermal power of the BARS reactor core could be influenced largely by thermal-hydraulic performance of the tight lattice fuel. The fuel is designed as triangular lattice whose rod gap is 1.3 mm. In such a tight lattice bundle, there are scarce critical power test data and critical power correlation applicable to critical power prediction.

Then, it was planed to three tests:

(i) Visualization test
(ii) High-pressure thermal-hydraulic test
(iii) CCFL (Counter Current Flow Limitation) test

5.1. Visualization test

The purpose of the visualization test is to investigate the boiling transition behavior in the narrow gap bundle. It is hard to visualize the two-phase flow in the rod bundle under BWR operating condition. Therefore, it is planed to perform the two single channel tests under atmospheric condition. One is the unheated test for measuring the physical quantities of the two-phase flow in narrow channel, for instance, liquid film thickness, velocity and so on.

The other is the heated test to visualize the liquid film behavior just before occurring the boiling transition. The heater rods are made of glass, whose surface is coated by $SnO2$. Because the boiling transition behavior will be investigated, these test sections have the same flow geometry.

5.2. High-pressure thermal-hydraulic test

5.2.1. Fundamental thermal hydraulic test

The purpose of the high-pressure thermal-hydraulic test is to make a database of the critical power performance of the tight lattice bundle whose rod gap is about 1.3 mm. In our previous study [18], we had carried out the critical power test with the tight lattice bundle and found that critical power correlation, developed by Arai [19], could be applied to predicting critical power of tight lattice bundle. However, the previous critical power test was performed with the longer heating length than that of BARS core in this paper. Therefore, the thermal-hydraulic tests were planned to enhance the thermal-hydraulic database for the tight lattice bundle, to verify the applicability of Arai's correlation to the BARS core, and to develop a more accurate correlation for predicting critical power. Moreover, the two-phase flow instability test, transient boiling transient test and pressure drop test were planned in order to design the BARS core from the thermal-hydraulic viewpoint.

Figure 22 shows the cross-sectional view of test assemblies. As shown in Fig. 22, it has been planned to fabricate two types of the test bundle. One is a 7-rod test bundle with a hexagonal channel box. The other is a 14-rod test bundle with a rectangular one. The purpose of the 7-rod bundle test is to survey the rod gap effect and heating length effect on the critical power and pressure drop. On the other hand, the purpose of the 14-rod bundle test is to check the critical power for various radial power distributions.

Figure 23 shows the system diagram of the test facility. This test facility is usually called BEST (Toshiba BWR Experimental Loop for Stability and Transient test). This loop is capable of testing under BWR operating conditions.

In our critical power test, pressure, inlet water tem perature and flow rate were set to the programmed levels first. Then, the bundle power was raised step by step in steps of a small magnitude. Critical power was defined as a power when the rod surface temperature jumped by 14 centigrade from the temperature under nucleate boiling conditions. Test conditi ons were planned to be: (1) Pressure: 5~8 Mpa; (2) Mass flux: 500 ~ 2000 kg/m2s; and (3) Inlet subcooling: 25~100 kJ/kg.

The critical power tests of the 7-rod test bundle completed. In this section, these typical test results, especially those concerning critical power, are reported.

The four types of test bundle were prepared for the 7-rod test. The test bundle consisted of the header rods, honeycomb-type spacers and hexagonal channel box. The stepped-cosine distribution was applied to the axial power distribution. The center rod peak distribution was also applied to radial power distribution.

The thickness and height of the spacer are 0.3 mm and 25 mm, respectively.

First, the results of the steady-state critical power test are mentioned. Figures 24, 25 and 26 show the dependence of the flow parameters on the critical power parameter. As the flow rate becomes larger, the critical power becomes larger. As the inlet subcooling becomes larger, the critical power becomes slightly larger. As the system pressure becomes higher, the critical power becomes lower. These flow parameter dependencies on critical power are similar to those of the conventional BWR fuel. The boiling transition was observed at only downstream edge of heating region, when the mass flux was lower than about 1000kg/m2/s. When the mass flux was higher than about 1000kg/m2/s, the boiling transition was observed at not only downstream edge but also just upstream of the first spacer from the downstream edge of heating region. The boiling transition did not occur at the highest heat flux region. Therefore, it was supposed that the boiling transition occurred due to liquid film dryout. The temperature behavior when boiling transition was observed was similar to that in the case of the conventional BWR fuel mock-up test.

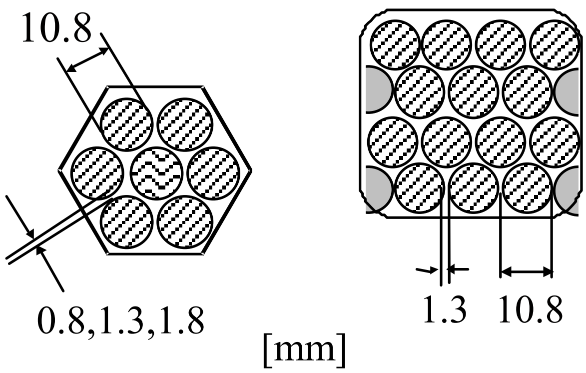


FIG. 22. Cross-sectional view of test assemblies.

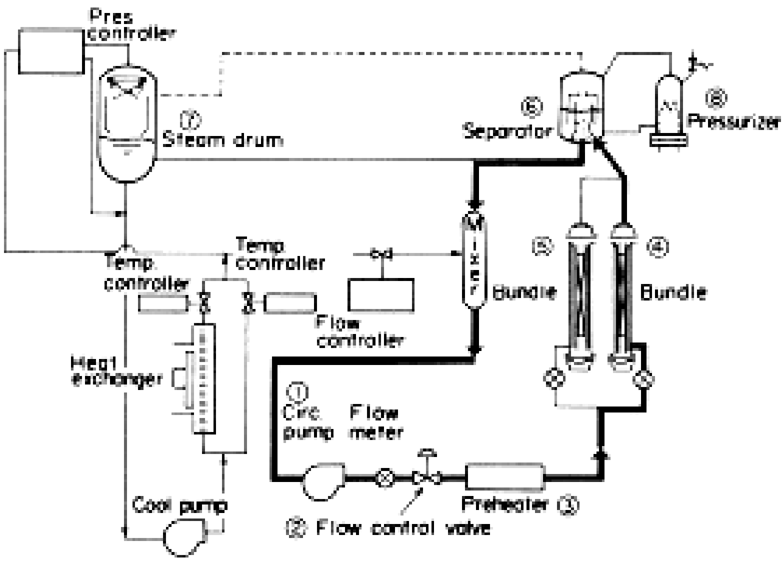


FIG. 23. Flow diagram of test facility (BEST loop).

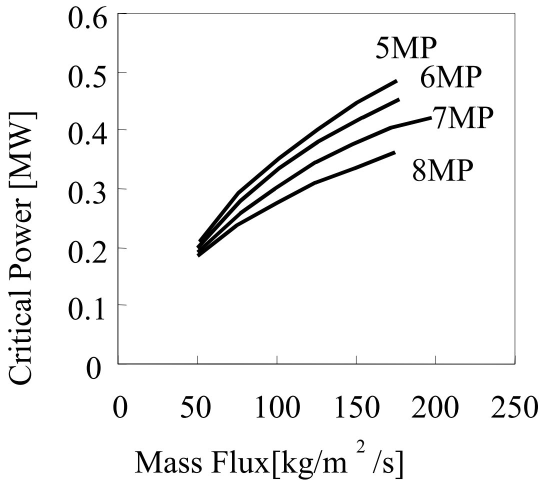


FIG. 24. Dependence of mass flux on critical power.

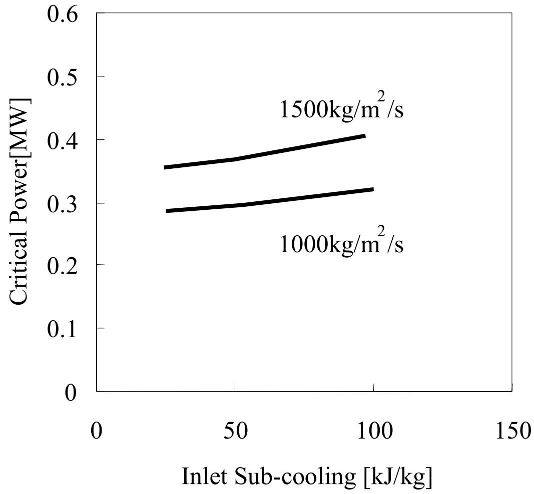


FIG. 25. Dependence of inlet subcooling on critical power.

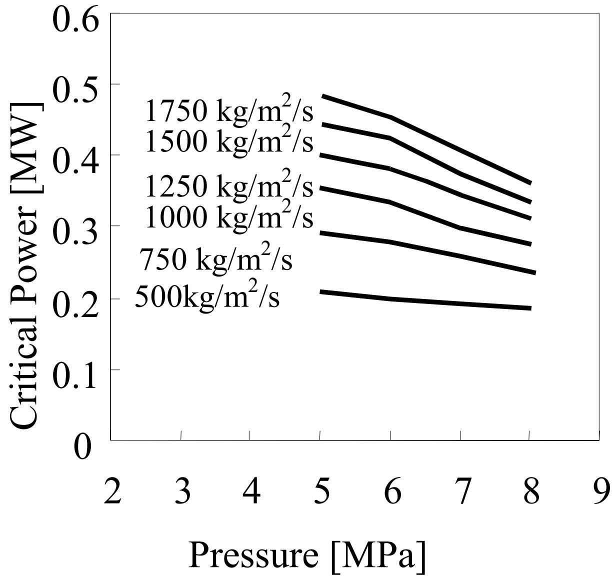


FIG. 26. Dependence of pressure on critical power.

Next, the rod gap effect on the critical power performance is mentioned. Figure 27 shows the mass flux dependence on the critical power for different rod gap cases. The critical power results of 1.8-mm gap test bundle were the highest of all under the same mass flux condition. Because the flow area of 1.8-mm gap test bundle was the largest for these test bundles, the total flow rate of 1.8-mm gap test bundle was also the largest under the same mass flux condition. Figure 28 shows the total flow rate dependence on the critical power for different rod-gap cases. The critical power results of 1.8-mm gap test bundle were also the highest of all under the same flow rate condition. This result means the critical power depends on the rod gap.

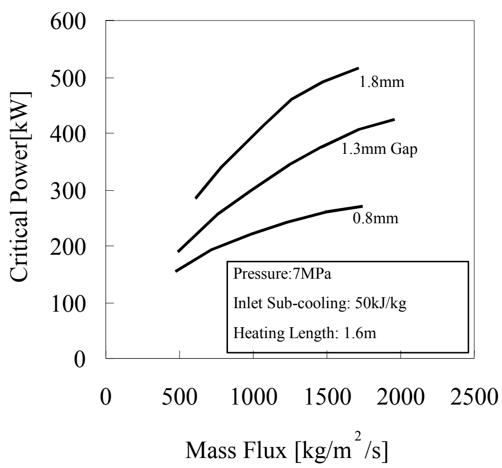


FIG. 27. Mass flux dependence on the critical power for different rod-gap cases.

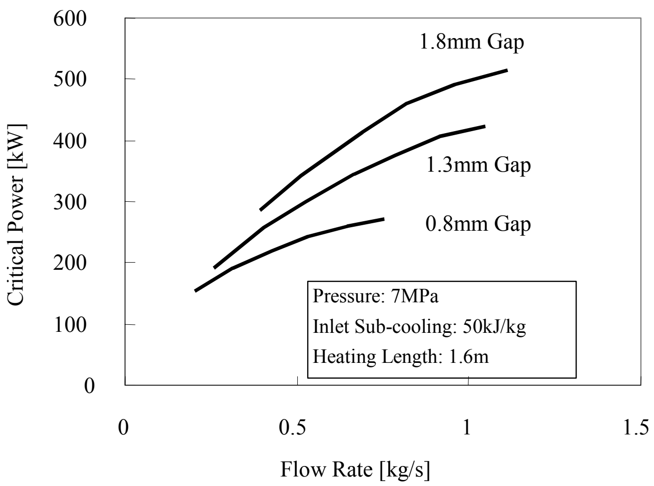


FIG. 28. Flow rate dependence on the critical power for different rod- gap cases.

Figure 29 shows the heating length effect on the critical power. Although the heating length becomes 0.8m from 1.6m, the critical power is larger than half of the critical power with heating length of 1.6m.

5.2.2. Critical power correlation

In our previous study (Yamamoto, 2002), it was mentioned that the critical power correlation, developed by Arai et al., was applicable to prediction of the critical power of tight lattice bundle. The Arai's correlation is based on the correlation developed by Biasi and Phillips [20], and based on the database by LeTouneau [21]. Arai's correlation is shown below.

G: Mass flux [kg/m2/s], Xc: Critical quality, LB: Boiling length [m], Ph: Heated perimeter [m], Pw: Wetted perimeter [m], Dh: Hydraulic diameter [m], Rf: Local peaking factor, hfg: Latent heat [J/kg], P: Pressure [Pa]

Then, Arai's correlation was applied to our new data. Figure 30 shows the measured critical power versus calculated critical power by Arai's correlation. Almost all the calculated critical power data for 0.8-mm and 1.3-mm gap test bundles were larger than the measured data. On the other hand, the calculated critical power data for 1.8-mm gap test bundle were the same as the measured data except for larger power data than 0.4 MW.

As a result, it was found that Arai's correlation had insufficient applicability to predict the critical power of the tight lattice bundle. In particular, the rod gap dependence on the critical power must be improved.

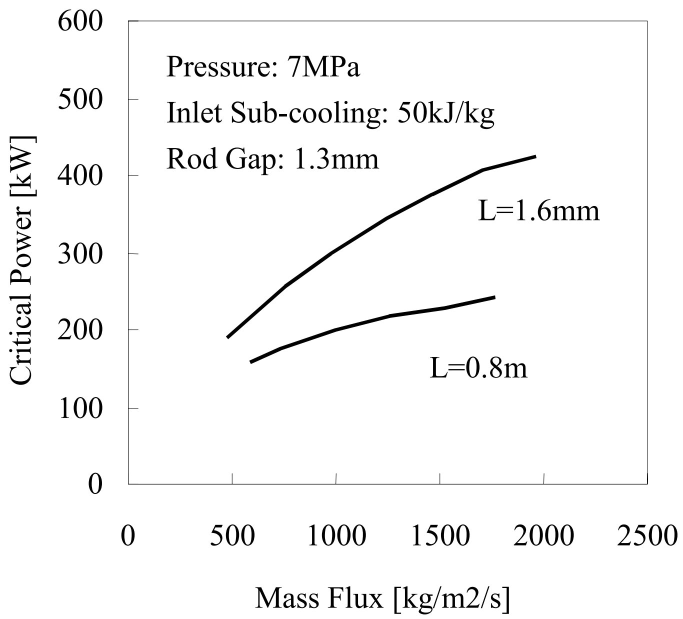


FIG. 29. Heating length (L) effect on the critical power.

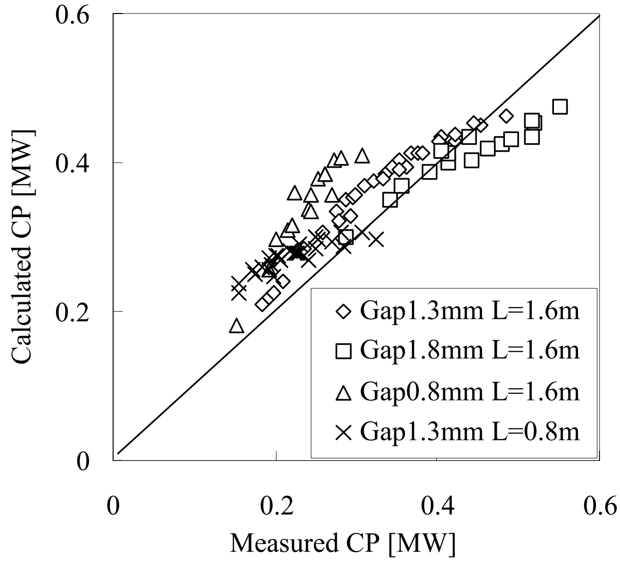


FIG. 30. Measured critical power versus calculated critical power with Arai's correlation.

To understand these results, the subchannel analysis was performed. The analysis showed that the mass flux at center subchannel decreased greatly downstream of the test bundle. On the other hand, the analysis showed that the mass flux at center subchannel did not decrease greatly downstream of the test bundle. In our previous test bundle, there were 6 dummy rods between channel tube and heater rods. It was supposed that these dummy rods contributed to unification of the coolant. These subchannel analysis results showed that the flow distribution had a large effect on the critical power performance in rod bundle.

Based on these results, it was found that the consideration of the flow distribution was necessary to predict the critical power accurately. Moreover, these results suggested that the critical power would become larger if the flow distribution were uniform.

The critical power tests using the 7-rod test bundle with narrow gap were performed in order to augment the database of the critical power performance applied for BARS core design. The following conclusions are obtained from this study.

(i) The dependencies of the flow parameter on the critical power for the tight lattice bundle are similar to those for the conventional BWR fuel bundle.
(ii) The calculated critical power by Arai's correlation does not agree with experimental data for 1.6-m length bundle, whereas good agreement was obtained for the previous test for 3.7-m length bundle. Therefore, further improvement of the critical power correlation is needed for the tight lattice bundle.

5.3. CCFL (Counter Current Flow Limitation) test

On the tight lattice bundle, because the flow area becomes narrower than that of the usual BWR fuel, it is worried that the little spray water falls into the core when LOCA (Loss of Coolant Accident) occurred. Therefore, the purpose of this test is to investigate the CCFL characteristic and to make a CCFL correlation for the tight lattice bundle. It is planed that the correlation based on this CCFL test is applied to the LOCA analysis. The conditions that the water is unable to fall down into the test bundle are defined CCFL condition. In CCFL test, the water sprays to the test bundle from the upper plenum. On the other hand, the vapor is blown into the test section under the test bundle.

6. CONCLUSIONS

BARS core design studies, critical assembly test study and thermal hydraulic test study have been done.

Cored design studies based on pin-cell geometry and bundle geometry have revealed the fundamental nuclear performance of BARS bundle. Streaming channel study has shown that the checker board location is more favorable than the layered location for negative conversion ratio.

Critical assembly tests based on modified conversion ratio method at NCA facility has been done. The experiment showed that nuclear design method on uranium core had adequate accuracy, because the conversion ratio and multiplication factor k* by experiment showed good agreement with the analysis.

Thermal hydraulic test study has been done as visualization test, high pressure thermal-hydraulic test. Visualization test has been planned to investigate the boiling transition behavior in the narrow gap bundle. High pressure thermal-hydraulic test has shown the thermal hydraulic performance and critical power for tight lattice. The flow parameter on critical power showed similar performance to those of convensional BWR fuel bundle. On the other hand, the calculated critical power did not showed good agreement with Arai's correlation. CCFL test study has been planned for tight lattice to prepare the CCFL correlation for LOCA analysis.

NOMENCLATURE

S(r): neutron production rate at position r

A(r): neutron absorption rate at position r

$(\nu\Sigma_f(r)\phi(r))_{25}$: production rate in ^{235}U

$(\nu\Sigma_f(r)\phi(r))_{28}$: production rate in ^{238}U

$(\Sigma_f(r)\phi(r))_{25}$: fission rate in ^{235}U

$(\Sigma_f(r)\phi(r))_{28}$: fission rate in ^{238}U

$(\Sigma_c(r)\phi(r))_{28}$: capture rate in ^{238}U

$(\Sigma_a(r)\phi(r))_{NF}$: absorption rate in Non Fuel

7. ACKNOWLEDGEMENTS

This project is funded by the Institute of Applied Energy (IAE), Ministry of Economy, Trade and Industry (METI), Japan.

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ADVANCED NUCLEAR ENERGY SYSTEMS FOR INHERENTLY PROTECTED PLUTONIUM PRODUCTION

M. SAITO, V. ARTISYUK, Y. PERYOGA, K. NIKITIN

Research Laboratory for Nuclear Reactors
Tokyo Institute of Technology
Tokyo, Japan

Abstract. The paper introduces an activity on Protected Plutonium Production (P^3-project) focusing the potential of light water reactor technology. The stress is placed on increasing the fraction of ^{238}Pu which attributes an essential protective measure to plutonium due to its elevated decay heat and neutron source from spontaneous fissions. Within the environment of light water reactors the ^{238}Pu production is stimulated by increased uranium enrichment and doping of Minor Actinides (MA), these options making the counteractive effects on reactor criticality. In this study their combination is encompassed with two boundary levels of uranium enrichment: 5% to stress the current commercial technology and 20% which is the upper limit in keeping uranium out of the direct usable weapon materials. The paper attempts at shaping the domain within the fuel burnup and MA doping to maximize ^{238}Pu production in both critical and subcritical operation modes.

1. INTRODUCTION

The efforts on assessing and elevating the proliferation resistant properties attributed to nuclear fuels were under scrutiny since the early stage of nuclear power development and, in the beginning of 80s, were well summarized in a comprehensive review on denaturing of fissile materials [1]. The principles of isotope denaturing presented in that report revealed critical mass as a limiting factor for uranium enrichment (no more than 20%) and intrinsic radiation of ^{238}Pu (natural alpha-decay and spontaneous fission) and ^{240}Pu (spontaneous fission) as a protective barrier for plutonium. Neptunium and americium from spent fuel of current reactors have been long recognized as a supply for plutonium denaturant and it is hardly worth selecting for reference any particular article on their recycling in fission reactors for this purpose. The IAEA regulations identify plutonium with ^{238}Pu fraction more than 80% as an exemption from safeguard [2]. This value, however, is to be considered as an upper limit and, as a matter of fact, any noticeable increase in ^{238}Pu content beyond 2% (characteristic of present day PWR with burnup of about 40 GWd/tHM) is referred to as an essential measure in improving plutonium proliferation resistance. For example the level of 5% was advertised in 1980 in Ref.[3] as one which is sufficient to cause melting the chemical explosive surrounding the plutonium sphere. The same fraction was stressed more than a decade later in Ref.[4] as a limit for available reprocessing technology and, for that reason, to be enough proliferation resistant to introduce the plutonium bearing fuels in developing countries. To summarize, it seems to be prudent to say that maximization of ^{238}Pu content towards 80% is a worth studying and this forms the main subject of the present paper.

In contrast to plutonium case, application of nearly 20% enriched uranium (henceforth referred to as *medium enriched uranium*) only recently received an increasing interest in exploring its proliferation resistant potential in light water reactors. Two distinctive approaches are being developed both aiming at enhancing the proliferation resistance in a once-through fuel cycle. One of them advocates multi-butch loading (on the annual base) of medium enriched uranium seed to drive subcritical thorium blanket with accumulation and burnup in situ ^{233}U denatured homogeneously by nonfissile ^{238}U [5,6]. This concept envisages also some fraction of 20% enriched uranium in thorium pins partly for enhancing the ^{235}U-^{236}U-^{237}Np transition chain to feed ^{238}Pu accumulation. Drivers by sophisticated shuffling scheme keep their reactivity up to about 70 GWd/tHM with final ^{238}Pu content of 7%, while thorium assemblies stay in the core as long as 10 yr to pursue the burnup of about 100 GWd/tHM thus resulting in ^{238}Pu fraction as high as 12-14%, [5,6]. This example provides a key to understanding the essence of non proliferation potential of uranium fuels with light water technology that is to be found in combined effect of long residence time and strengthening the ^{235}U-^{236}U-^{237}Np denaturing chain both backed up by 20% enrichment.

Another, a rather straightforward option, focuses the maximization of burnup for 20% enriched uranium oxide (UOX) [7]. In Ref.[8] it was shown that burnup of 170 GWd/tHM can be achieved in a sequence of three irradiation cycles in light water reactors (without intermediate radiochemical reprocessing) with different moderator-to-fuel ratio, and at the end of the final irradiation cycle the ^{238}Pu fraction is 12% that is similar to denaturing in a seed-blanket approach [5,6]. These burnup and degree of proliferation resistance seems to be maximum what can be gain on with critical operation in LWRs fed by proliferation resistant UOX. Further plutonium denaturing obviously preassumes the MA doping. The concept of long-life reactor core with U-Np fuel is advertised in Ref.[9], where equilibrium recycling mode with self-generated neptunium (1.3% at. of total HM) and plutonium (3.2%) reveals, in a one-butch irradiation, fuel burnup of 140 GWd/tHM with final equilibrium plutonium vector enriched up to 46% with ^{238}Pu. Partial effect of transputonium (Am+Cm) doping to 20% enriched UOX burnup was investigated for once through irradiation, exhibiting 100 GWd/tHM burnup and plutonium vector with ^{238}Pu fraction of 20% [10].

It is worthwhile to emphasize that in all the above mentioned scenarios [5-10] the focus is placed on increasing the fuel burnup without compromising the reactivity issues beyond the margins inherent for current LWRs. So, plutonium vector is illustratively shown just as it is at the discharge. The present paper intends to extend the domain of searched parameters with more deep insight into interrelations of fuel reactivity, burnup and dynamics of plutonium vector. The paper begins with analysis of the effect of enrichment on ^{238}Pu dynamics in a typical LWR configuration, than considers the doping of various ^{238}Pu predecessors to conventional (5% enrichment) and to 20% enriched UOX fuel. Three options of doping are considered. Among them are doping of pure ^{237}Np, doping of transplutonium elements (Am-Cm mixture) and doping of MA (Np-Am-Cm mixture), all in appropriate oxide form.

2. PRELIMINARY NEUTRONICS CONSIDERATION

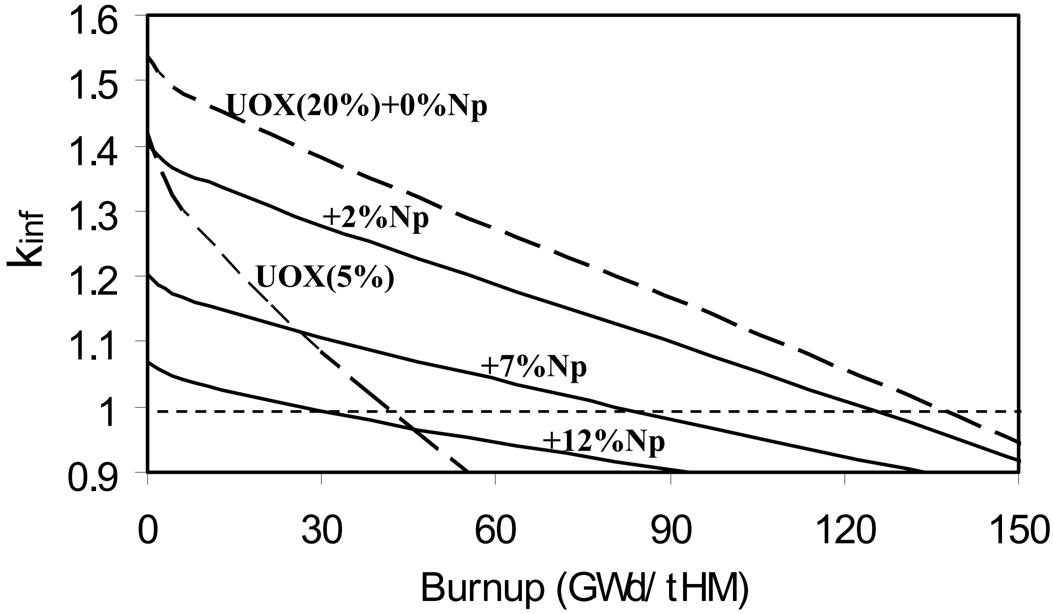
The fuel depletion and criticality calculations were performed in a unit pin-cell model with the help of the SCALE4.4 code system [11].The SAS2H control module of the SCALE4.4 coupled with 44-group cross-section library generated from ENDF/B-V [12] was applied for burnup calculations. The average cross sections derived from a transport analysis at each time burnup step are used in a point depletion computation (via ORIGEN-S) to determine the fuel composition for the next run.

Table I gives a summary of essential data for cell specification. Note that pitch is not specified here since it is used later as a free parameter to encompass the neutronics responses to a wide variation in spectrum conditions. Another distinctive point is the choice of cladding material that should challenge the variety of irradiation times which might significantly exceed that of conventional PWR in the case of medium enriched uranium. To secure the integrity of fuel pins under these conditions, fuel porosity of 15% and stainless steel cladding were adopted in calculational model in contrast to 5% and zircalloy for conventional UOX, correspondingly.

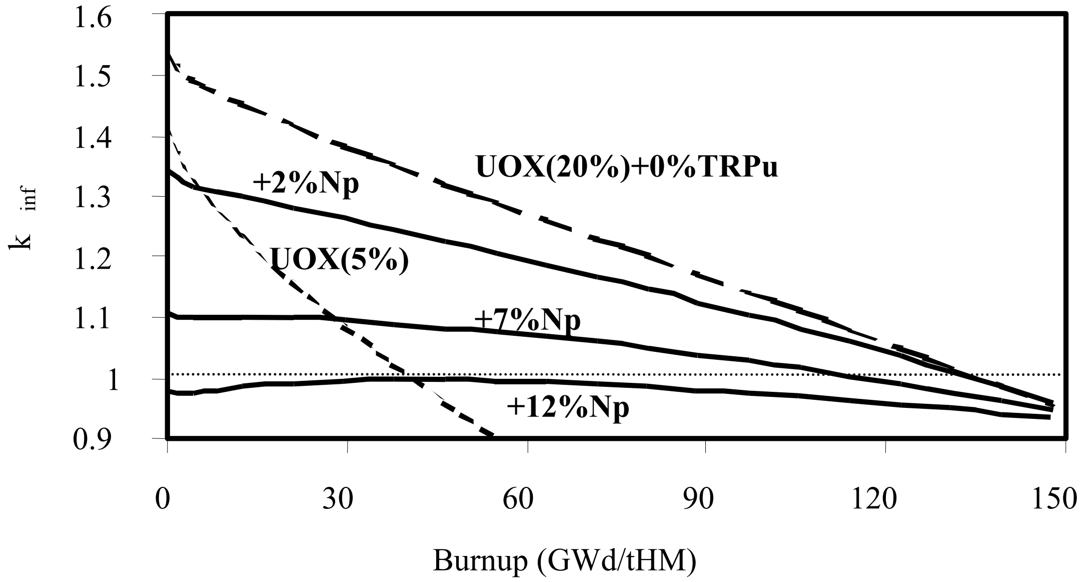
Figure 1 gives an illustrative scope of essential neutronics that encompasses burnup-dependent criticality (k_{inf}) for two boundary enrichment levels for typical PWR configuration (a pitch-to-diameter (P/D) ratio is fixed at 1.4) and gives an illustration of MA effect on the core performance for the case of medium enriched uranium. Standard case with conventional fuel (enrichment 5%) is also shown for comparison. Medium enriched uranium extends the fuel burnup to about 140 GWd/tHM that gives the maximum potential in developing the long-life reactor core based on proliferation resistant uranium fuels at given P/D confioguration. The MA doping is illustrated by Np addition and addition of transplutonium isotopes separately to reflect possible variation in the fuel cycle modes. A mixture of transplutonium isotopes is taken in a proportion between them as it appears in the PWR fuel with initial enrichment of 5% burnt up to 50 GWd/tHM and cooled 1 yr after discharge. In general MA doping shrinks the area of critical performance by reducing the initial criticality and the final burnup value (assumed at k_{inf} =1), though for the same mass addition transplutonium doping markedly reduces reactivity swings compared to Np effect. For conventional fuel with 5% enrichment doping makes a qualitatively similar effect and even small MA addition could make the critical operation impossible. Note, that combined effect of Np together with transplutonium elements give similar behavioural patterns. The doping effects on ^{238}Pu accumulation in more details are discussed in subsequent sections.

Table I. Cell specification assumed for calculational model

Fuel radius, cm	0.4095
Cladding radius, cm	0.475
Fuel density, g/cm^3	9.316
Fuel temperature, ^{o}K	933.15
Water temperature, ^{o}K	579.45



a) Effect of Np doping



b) Effect of trasplutonium (TRPu)doping

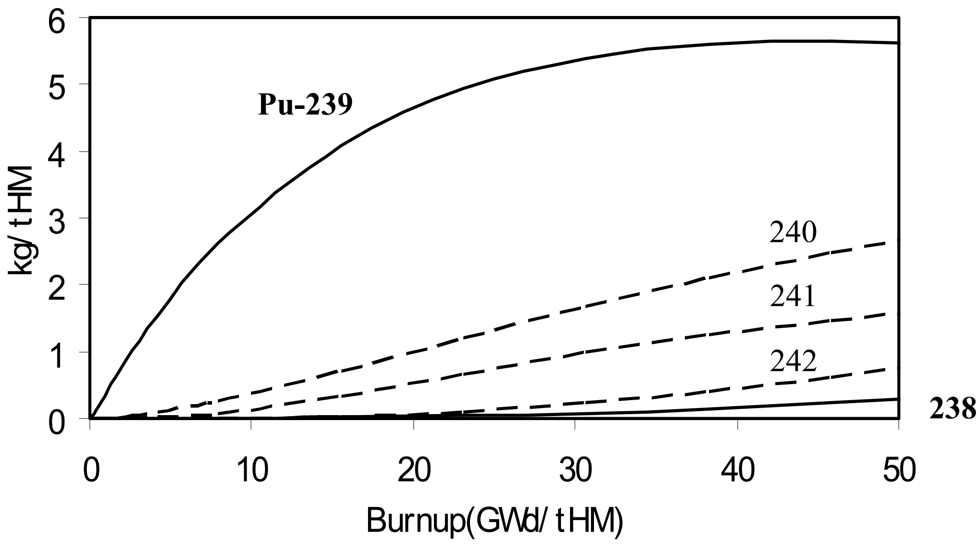
FIG. 1. Burnup of OUX fuels doped with neptunium and transplutonium elements (P/D=1.4).

3. EFFECT OF URANIUM ENRICHMENT ON ^{238}PU ACCUMULATION

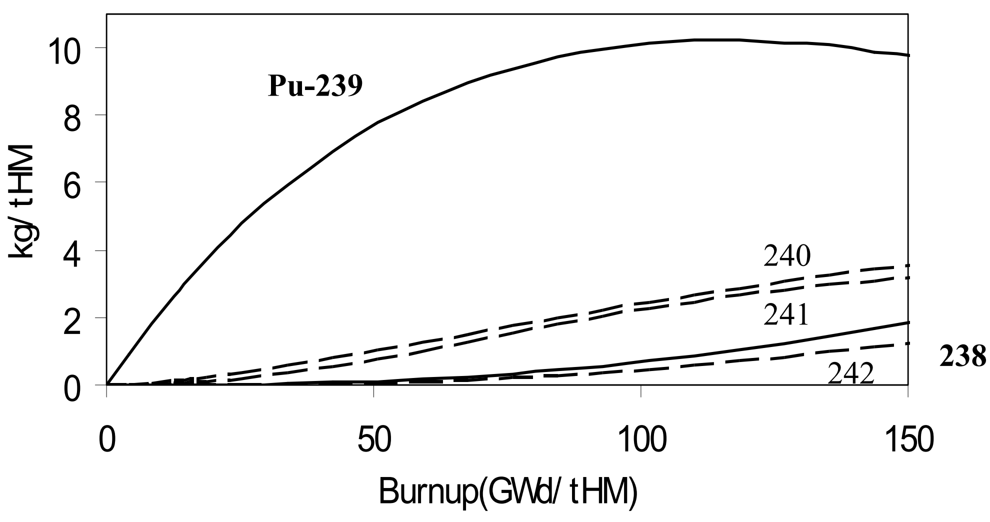
It is matter of fact, that ^{235}U is the primary source of ^{238}Pu accumulation through three-step capture reaction in conventional reactors

$$^{235}U \rightarrow ^{236}U \rightarrow ^{237}U(\beta)\ ^{237}Np \rightarrow ^{238}Np(\beta)\ ^{238}Pu. \quad (1)$$

Figure 2 shows the dynamic of accumulation of plutonium isotopes for two reference enrichment levels. Being expressed in mass units, the burnup dependent isotopic vector manifests that for higher enrichment the total plutonium accumulation is noticeably increased. This is a well known phenomenon that reflects the fact of enlargement of ^{238}U involvement in forming the neutron population in a hard spectrum.



a) 5% enriched UOX fuel



a) 20% enerichedUOX fuel

FIG. 2. Dynamic of Pu accumulation in UOX fuels (P/D=1.4).

Example of spectrum distortion, particularly the low energy part, is shown in Figure 3 for boundary uranium options. It is so hard in the case of medium enrichment that, in the course of the fuel burnup, it exhibits noticeable softening, the trend being opposite to the low enrichment case. Needless to say, doping of MA would make the spectrum distortion even more pronounced, since as it follows from Fig.1 it works mainly as a burnable absorber. So, the reference fixed P/D =1.4 is taken here merely for illustration and should be definitely adjusted to secure the reactor safety. Notsurprisingly, illustrated by Fig. 2 the irradiation of low and medium enriched fuels show the same mechanism. Domination of ^{238}Pu portion over the ^{242}Pu at relatively low burnups reflects the fact of proportionality of ^{238}Pu accumulation to ^{235}U concentration.

Burnup-dependent plutonium profile is given in Figure 4 which encompasses both critical operation and subcritical one stopped at k_{inf} =0.7 estimated to be as an energy breakeven condition for accelerator-driven system (ADS) with lead target irradiated by 1.6 GeV protons (about 50 neutrons are produced per one incident proton). For criticality less than this, energy balance in the ADS is negative. Irradiation of conventional fuel reveals saturation of total plutonium accumulation and peak of ^{238}Pu fraction around 100 GD/tHM. Further irradiation is hardly makes sense. For the case of 20% enriched uranium, this situation occurs at about 250 GWd/tHM which is also a the border line from the view point of energy balance in ADS. This gives some flexibility in shaping the plutonium vector with respect to subcritical fuel burnup and total plutonium output and its proliferation resistant qualities. It is worthwhile to note that energy supplied to accelerator within this burnup interval might be straightforwardly used as a cost of plutonium protection.

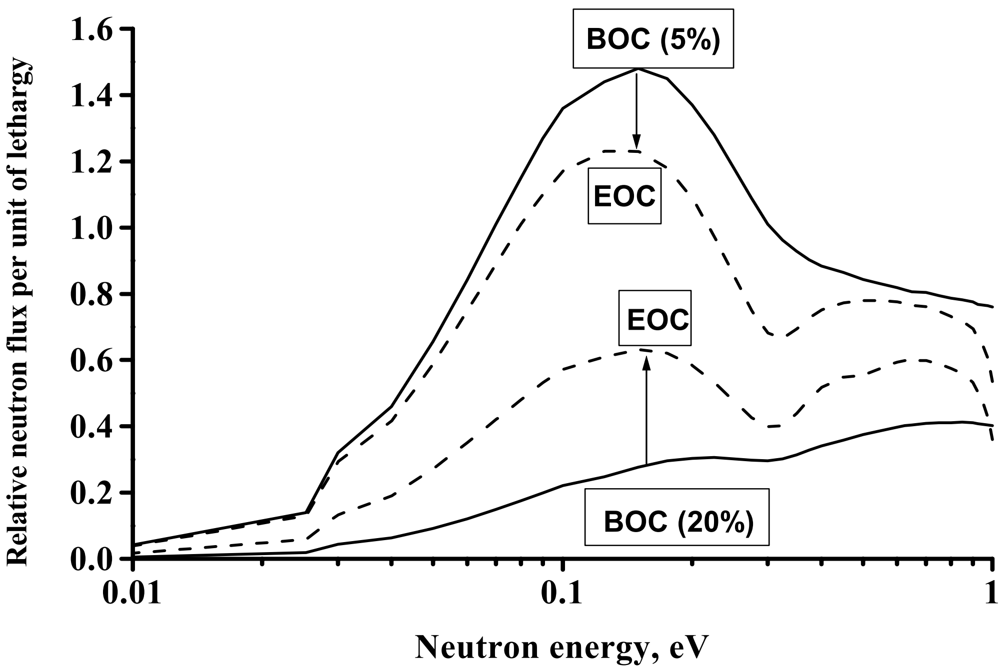


FIG. 3. Neutron spectrum for conventional and medium enriched fuel.

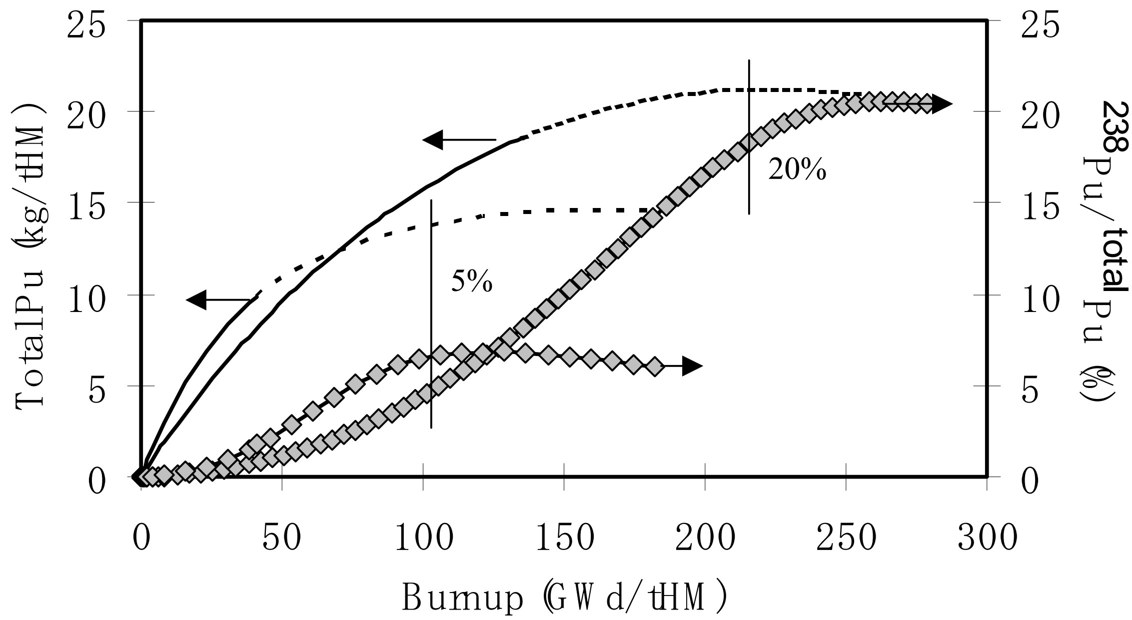


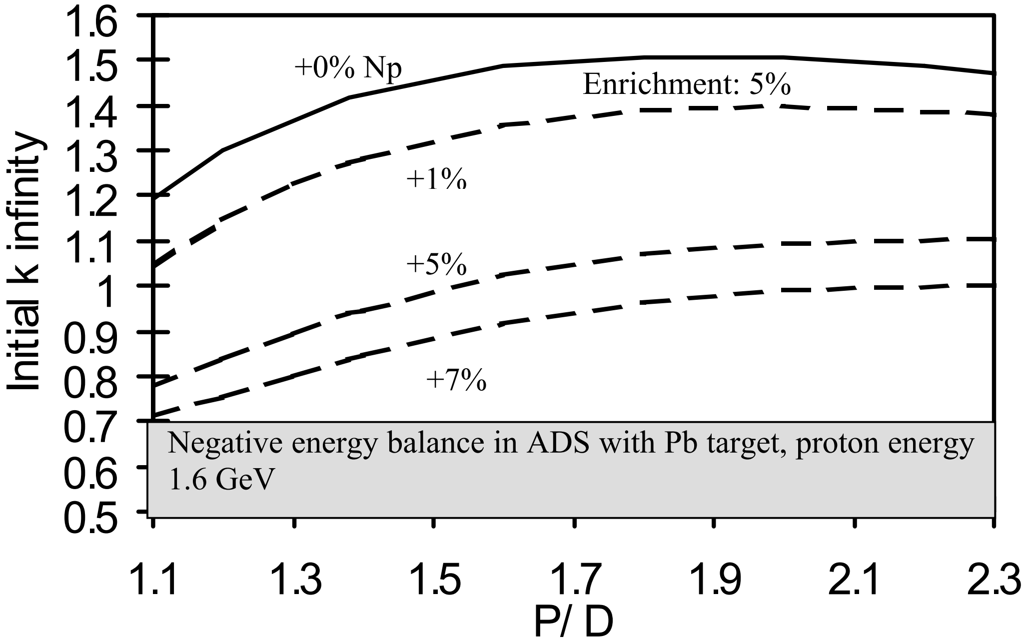
FIG. 4. Burnup dependent ^{238}Pu fraction for various enrichment levels (solid - critical; dotted lines subcritical operation stopped at k_{inf} =0.7).

4. EFFECT OF MINOR ACTINIDE DOPING ON ^{238}PU ACCUMULATION

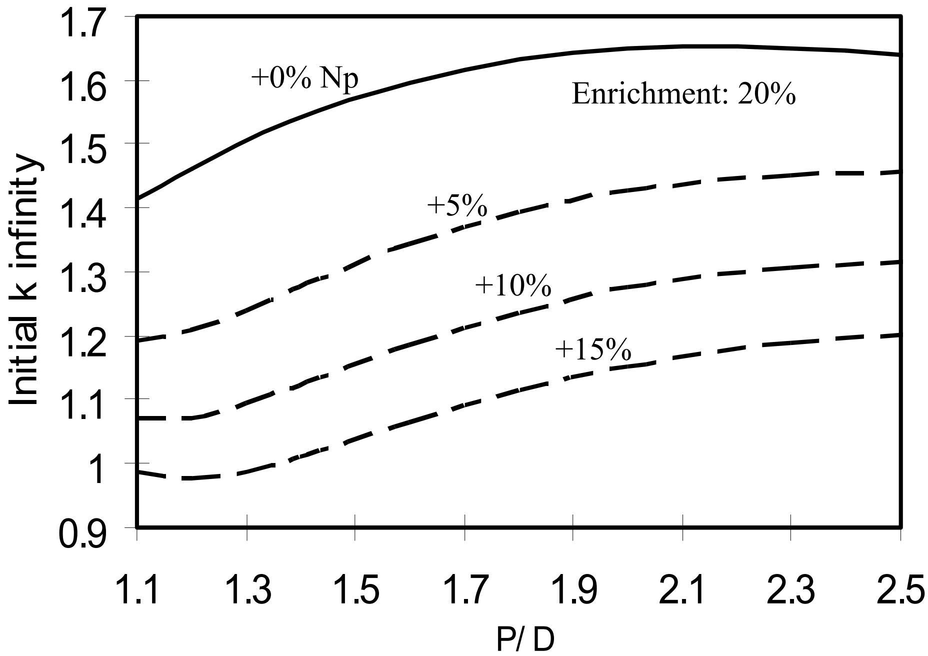
The subcritical operation mode appears to be an essential for production of proliferation resistant plutonium. So, in the subsequent section a consideration is given to identify the spectral conditions at which minor actinide doping brings the fuel reactivity below critical. It is searched for both conventional and medium enriched UOX through a broad P/D interval) and after that some representative cases are selected to demonstrate the peculiarities of isotopic vector behaviour.

4.1. Effect of neptunium doping

As shown in Figure 5a, Np doping of more than 5% to conventional fuel makes the PWR core subcritical at any feasible P/D ratio. In this situation it is energy balance of ADS that becomes a criterion in selecting the maximum doping. Dashed area in Fig. 5a borders the domain of negative energy balance of ADS operation with pointed ADS beam/target parameters. It seems that doping of 5% is the maximum that can assure nearly critical operation (k_{inf} >0.9) without drastic reduction in the core power density, P/D ratio adjustment being from 1.4 to 1.6. Within this interval highly enriched uranium permits the doping of 15% with no significant losing in positive initial reactivity (Figure 5b) what is essential since increased enrichment in fact is a cost of avoiding subcritical operation.



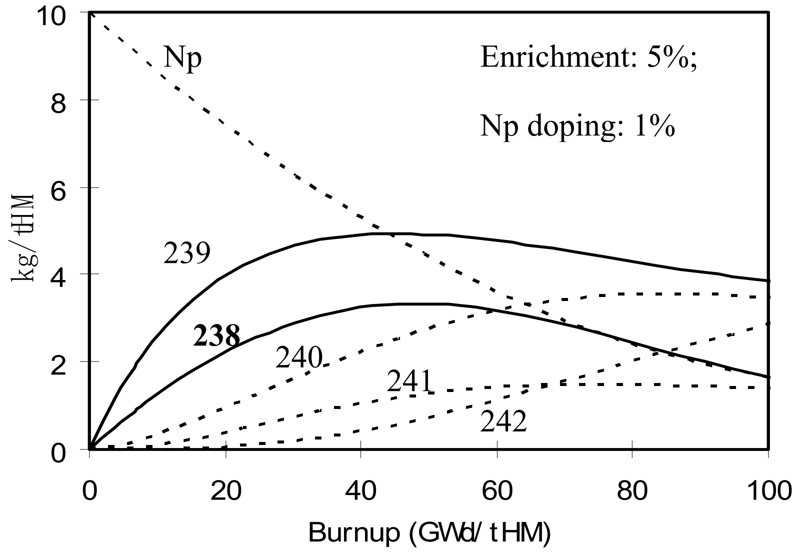
a) Np doping to 5%enriched UOX



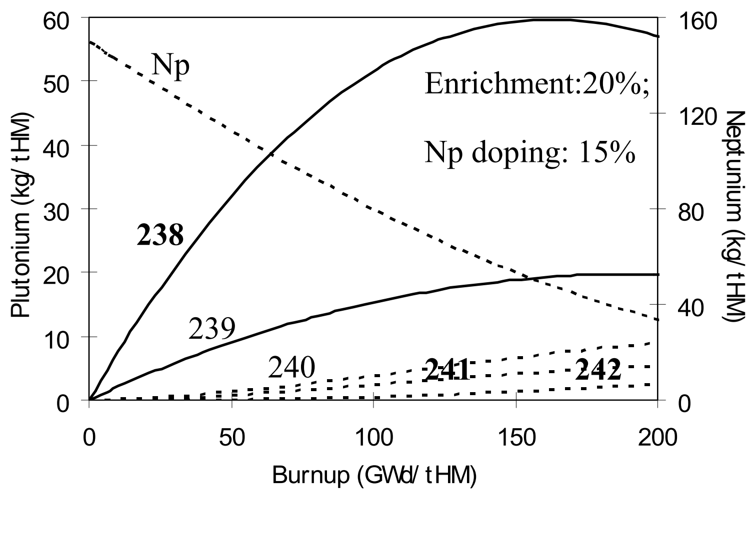
b) Np doping to 20% enriched UOX

FIG. 5. Initial criticality of Np doped UOX fuel in various spectra.

The isotope ^{237}Np is by two steps in neutron capture transition chain is closer to ^{238}Pu than ^{235}U and proportionally to its doping quantity the total plutonium output increases compared to undoped UOX as follows from comparison of Fig. 2 and Fig. 6. For example at the burnup level of 100 GWd/tHM for medium-enriched case, 15% Np doping increases the total mass output by factor 4. For conventional fuel the doping even of 1% makes the fraction of ^{238}Pu dominating over all the other isotopes except ^{239}Pu (Fig. 6a). Notsurpisingly, the higher is the Np fraction the faster ^{238}Pu accumulation proceeds. For reference case of 15% Np doping to 20% enriched fuel the rate of ^{238}Pu accumulation exceeds that of ^{239}Pu from the starting of irradiation.



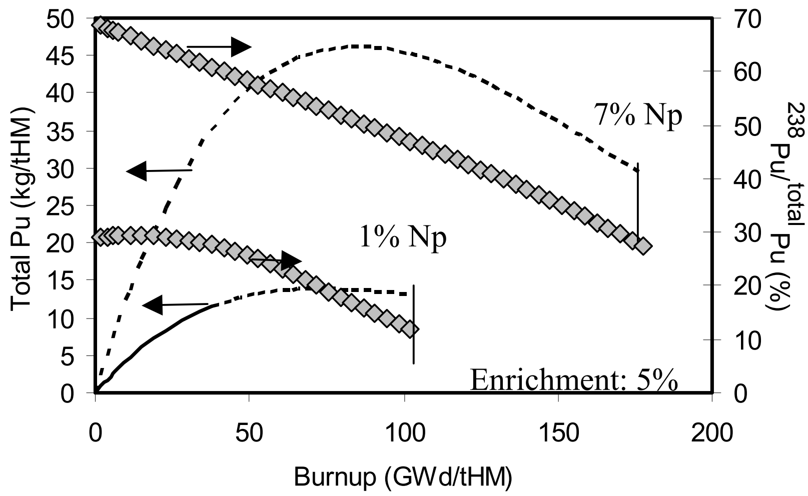
a) 5% enriched UOX fuel doped with 1%Np



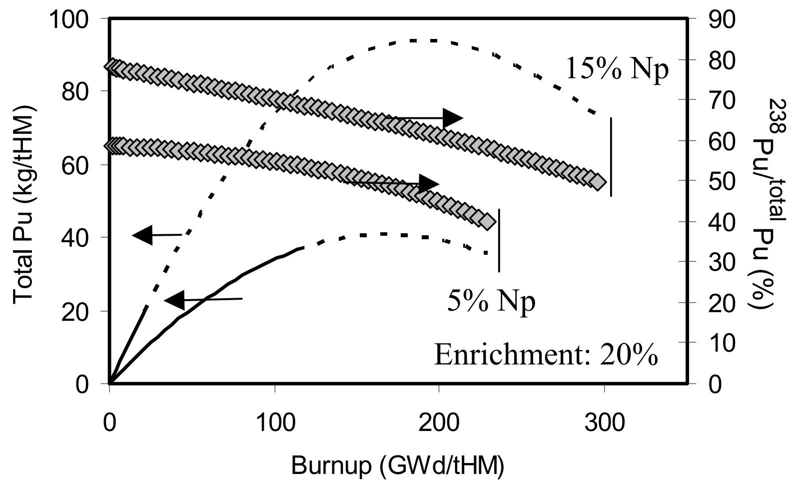
b) 20% enriched UOX fuel doped with 15%Np

FIG. 6. Dynamic of Pu vector in Np doped UOX fuels (P/D=1.6).

Thus, Np doping helps to protect plutonium produced even at small burnups. The level of this protection is governed exclusively by Np concentration and reduces in the course of fuel irradiation. Dynamic of ^{238}Pu fraction is well represented by almost monotonically declining curves as it is shown in Figure 7. The slope of the curves is somewhat less within the margin of critical operation for both conventional and medium enriched fuels. Np doping oriented for subcritical mode obviously gives an advantage in both quality of protection and mass production, however to which extent subcritical mode could keep this attribute depends upon energy balance in accelerator-driven system. This issue is not touched on in the present paper. The only conclusion can be made here is that from the view-point of production of proliferation resistant plutonium, it hardly makes sense to extend the burnup beyond the value at which peak of plutonium accumulation is approached.



a) Np doping to 5% enriched UOX



b) Np doping to20% enriched UOX fuel

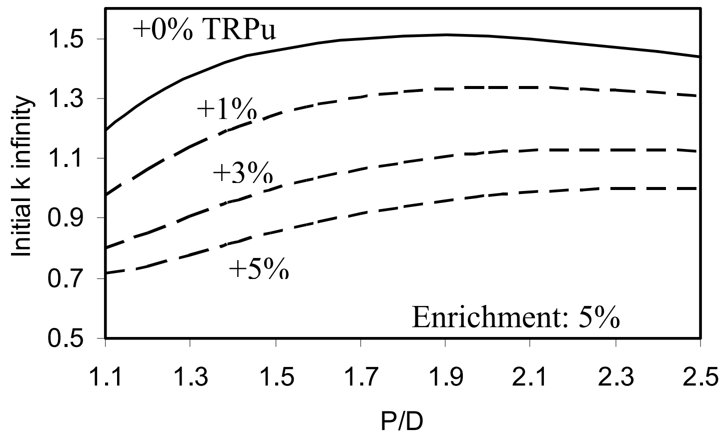
FIG. 7. Burnup dependent plutonium profile sNp doped UOX fuels(P/D=1.6, solid- critical; dotted lines -subcritical mode stopped at k_{inf}=0.7).

4.2. Effect of doping of transplutonium elements

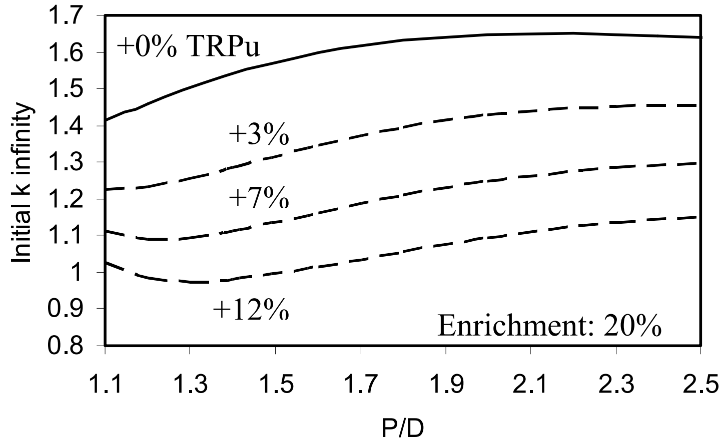
Transplutonium isotopes produce ^{238}Pu through neutron capture in ^{241}Am followed by two decays:

$$^{241}\text{Am}\rightarrow{}^{242}\text{Am}(\beta)\,{}^{242}\text{Cm}(\alpha)\,{}^{238}\text{Pu}. \tag{2}$$

As vividly shown in Figure 8, their doping reduces initial criticality which can be partly restored by adjusting P/D ratio. Following the logic assumed in the previous section, four reference cases are selected to feature plutonium behaviour to stress critical and subcritical operation for two types of UOX fuels. These are 1 and 5% dopings to conventional fuel and 3 and 12% to medium enriched fuel, correspondingly.



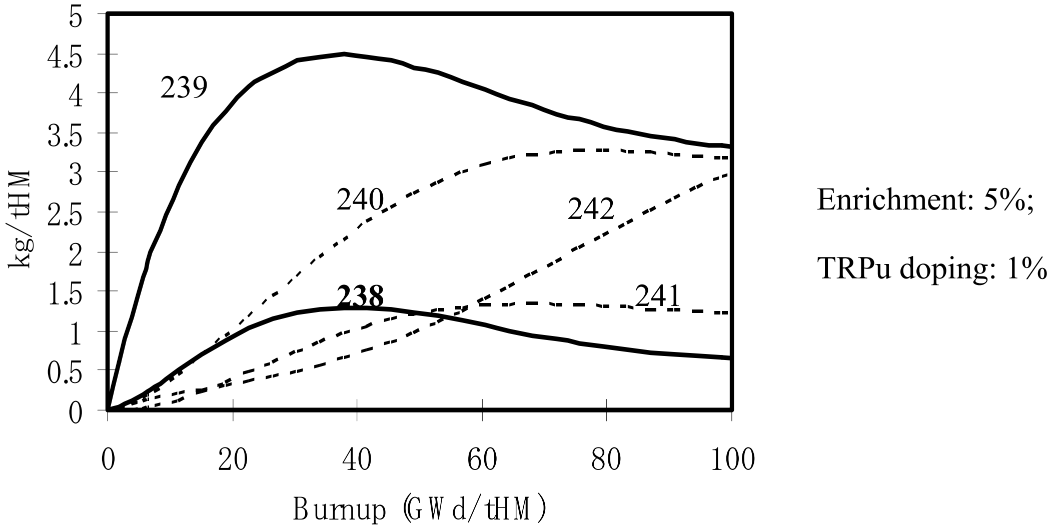
a) TRPu doping to 5% enriched UOX



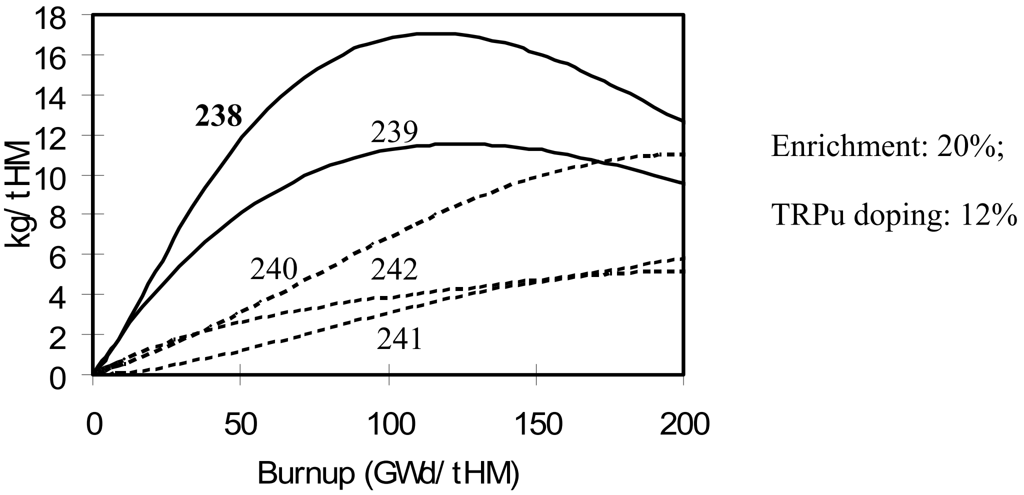
b) TRPu doping to20% enriched UOX fuel

FIG. 8. Initial criticality of UOX doped with transplutonium (TrPu) elements fuel in various spectra.

Compared to Np case, transplutonium doping gives less pronounced effect on increasing the fraction of ^{238}Pu (Figure 9) as well as on total plutonium generation (Fig. 10). Relative role of ^{240}Pu and ^{242}Pu becomes bigger due to relatively large content of ^{242}Am (electron capture transition to ^{242}Pu) and alpha decay of produced ^{246}Cm. Peaks of ^{239}Pu accumulation are observed al low burnup values (partly this is because weakening the transition chain from ^{238}Pu to ^{239}Pu) and degradation of ^{238}Pu fraction is rather pronounced. Generally, transplutonium doping gives a stronger effect on criticality than Np at the beginning and at the end of irradiation. For the same mass doping, reduction of initial criticality more essential (Fig. 1) as well as approaching the reference negative energy balance in subcritical mode that occurs at lower burnup values (Figure 10). However at intermediate burnup values the effect is more favourable for critical operation, since reactivity swing is less.

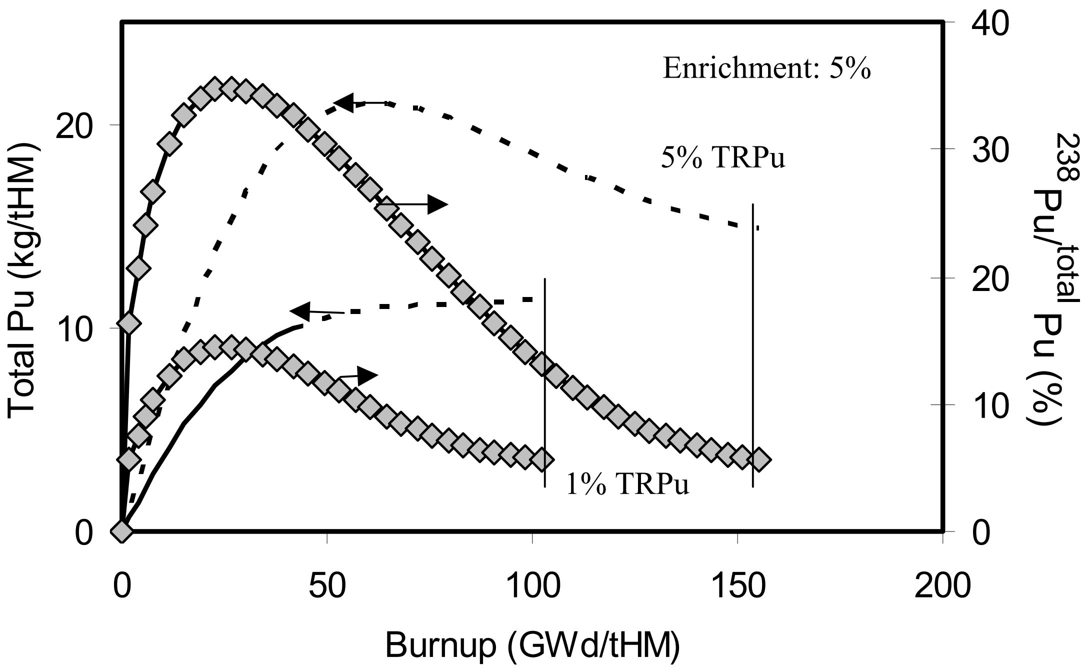


a)1% TRPu doping to 5% enriched UOX

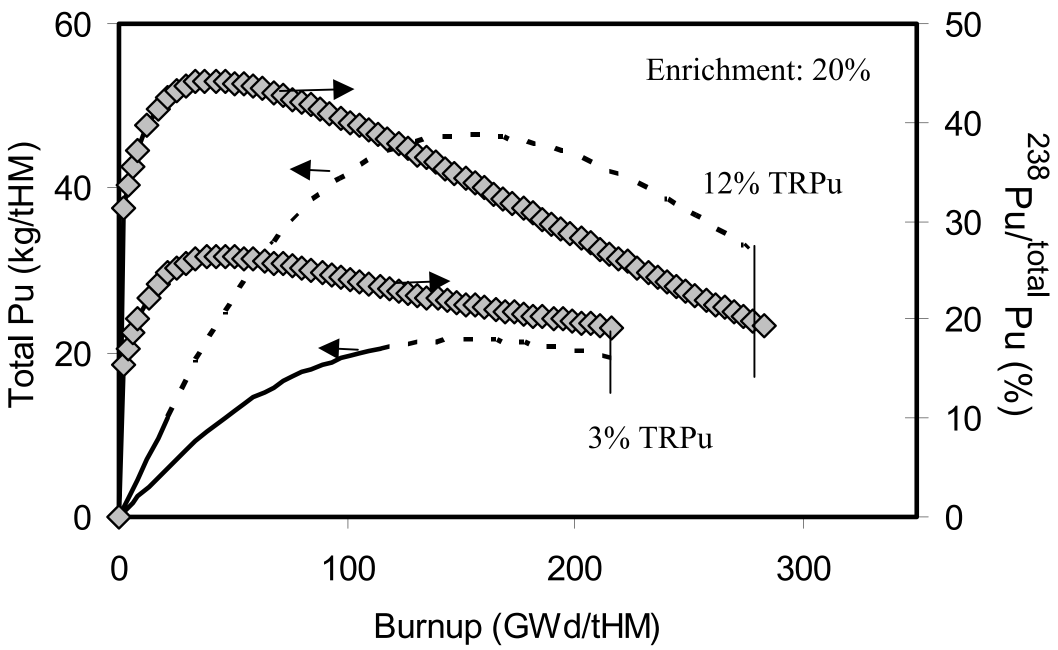


b) 12% TRPu doping to20% enriched UOX fuel

FIG. 9. Dynamic of Pu accumulation in UOX fuels doped with transplutonium (TRPu) isotopes (P/D=1.6).



a) TRPu doping to 5% enriched UOX

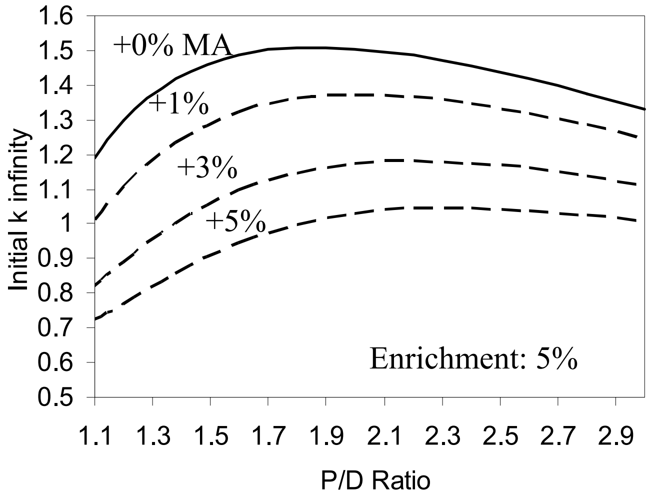


b) TRPu doping to20% enriched UOX fuel

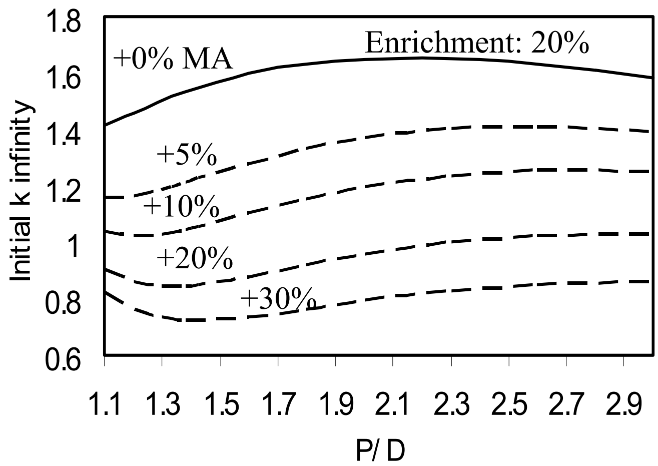
FIG. 10. Burnup dependent plutonium profiles for transplutonium doping (P/D=1.6, solid-critical; dotted lines -subcritical mode stopped at k_{inf}=0.7).

4.3. *Combined doping of minor actinides*

This section deals with combined effect of neptunium and transplutonium isotopes on characteristics of plutonium protection. Their mixture is taken in a proportion between them as it appears in the PWR fuel with initial enrichment of 5% burnt up to 50 GWd/tHM and cooled 1 yr after discharge. In mixture of discharged minor actinides, Np content is about 50%, so combined effect of MA as expected will bring the initial criticality for conventional fuel into subcritical domain at more than 1% of doping minor actinide mixture for standard LWR configuration with P/D=1.4, as seen from Fig. 11a. For medium enriched uranium case core becomes subcritical at more than 10% MA doping (Fig.11b). Burnup dependent criticality behaviour for medium enriched uranium is illustrated in Figure 12. It appears that 30% MA doping is close to maximum level that medium enriched uranium could accept in subcritical operation.



a) MA doping to 5% enriched UOX



b) MA doping to20% enriched UOX fuel

FIG. 11. Initial criticality of UOX doped with transplutonium (MA) elements fuel in various spectra.

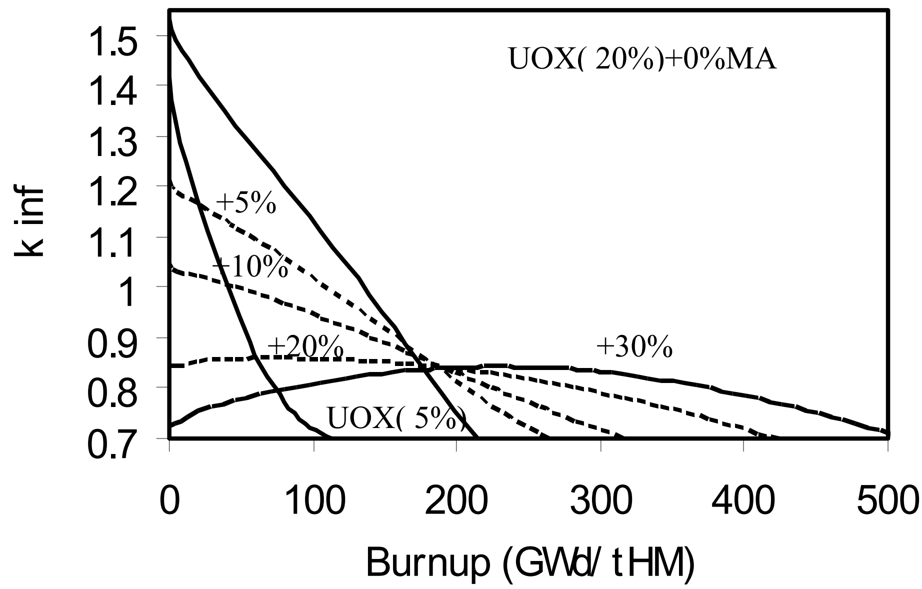
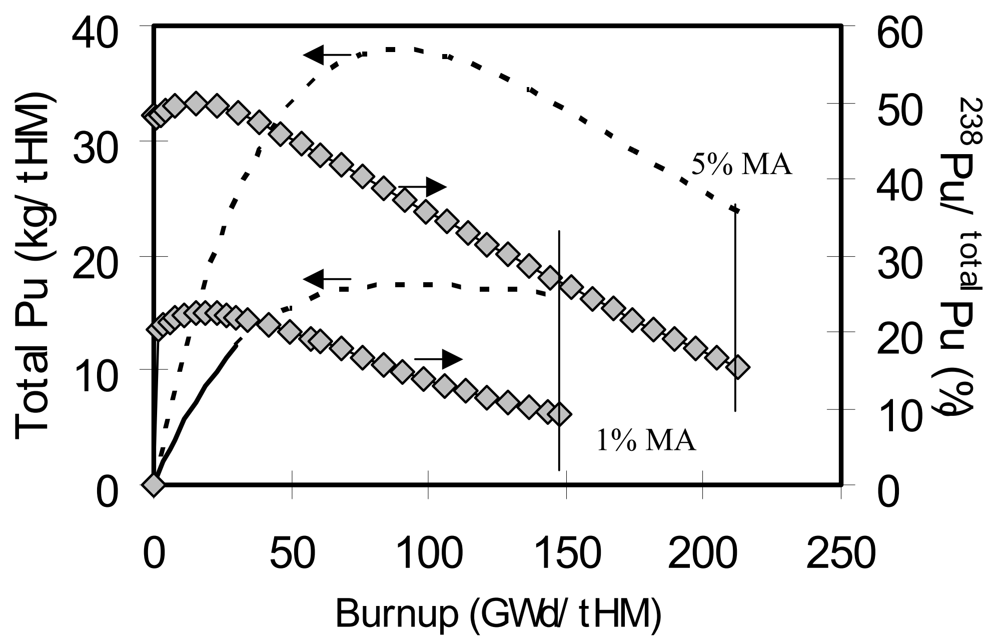
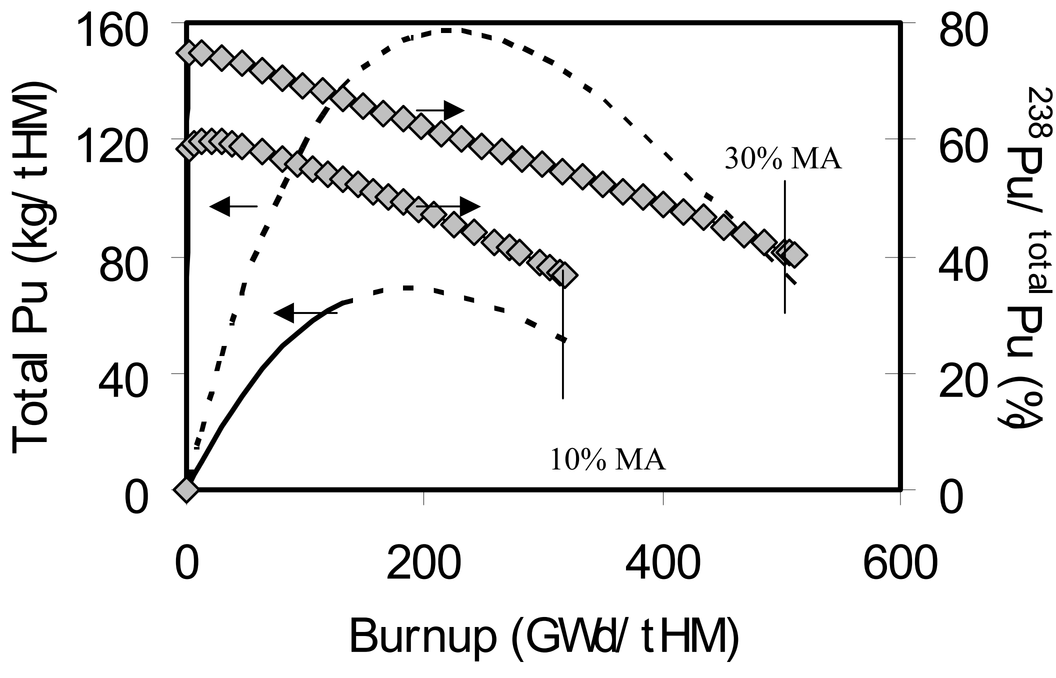


FIG. 12. Burnup of OUX fuels doped with minor actinides (MA) (P/D=1.4).

Due to large Np fraction, ^{238}Pu fraction is increased at low burnups compared to transplutonium doping, though obviously this effect is less than with pure Np doping. Illustrative patterns of plutonium accumulation are shown in Figure 13. They are well conceivable from the discussion given in the preceding sections.



a) MA doping to 5% enriched UOX



b) MA doping to 20% enriched UOX

FIG. 13. Burnup dependent plutonium profiles for MAdoping (P/D=1.4, solid- critical; dotted lines -subcritical mode stopped at k_{inf}=0.7).

5. CONCLUDING REMARKS

This paper considers doping of minor actinides to uranium oxide fuels as a powerful measure to improve the proliferation resistance of plutonium accumulated in the environment of light water reactors. The focus is placed on mechanism of increasing the fraction of ^{238}Pu which due to its high decay heat (560 W/kg) impairs the manufacturing of nuclear explosive. Two options of uranium enrichment are considered to highlight the technological cost of such a protection. The first is based on the current commercial technology with 5% of ^{235}U fraction. Another option is medium enriched uranium (20% which is the maximum to keep uranium out of the direct usable weapon material). Doping was varied in large interval to demonstrate peculiarities of plutonium generation in both critical and subcritical operation modes. Maximum fraction of ^{238}Pu inherently achievable with conventional fuel under critical operation is at the level 20%. Large fraction of their doping to medium enriched uranium supported by subcritical operation can increase this value to the level of more than 50%. Thus the paper quantitative outlines the domain of subcriticality, burnup values and ^{238}Pu fraction as well as total plutonium output for future optimization studies within the project on inherently Protected Plutonium Production (P^3-project) which aims at creating nuclear fuel cycle more flexible with respect to plutonium storage in view of its enhanced proliferation resistance. The proposed fuel cycle is sketched in Figure 14. Its main feature concerns with treatment of minor actinides. Instead of their geological disposal or just their burning through fissioning, it is conceivable to start production of proliferation resistant for future use. It would open the possibility for plutonium to be an energy treasure laid by present society as a message and gift to next generations.

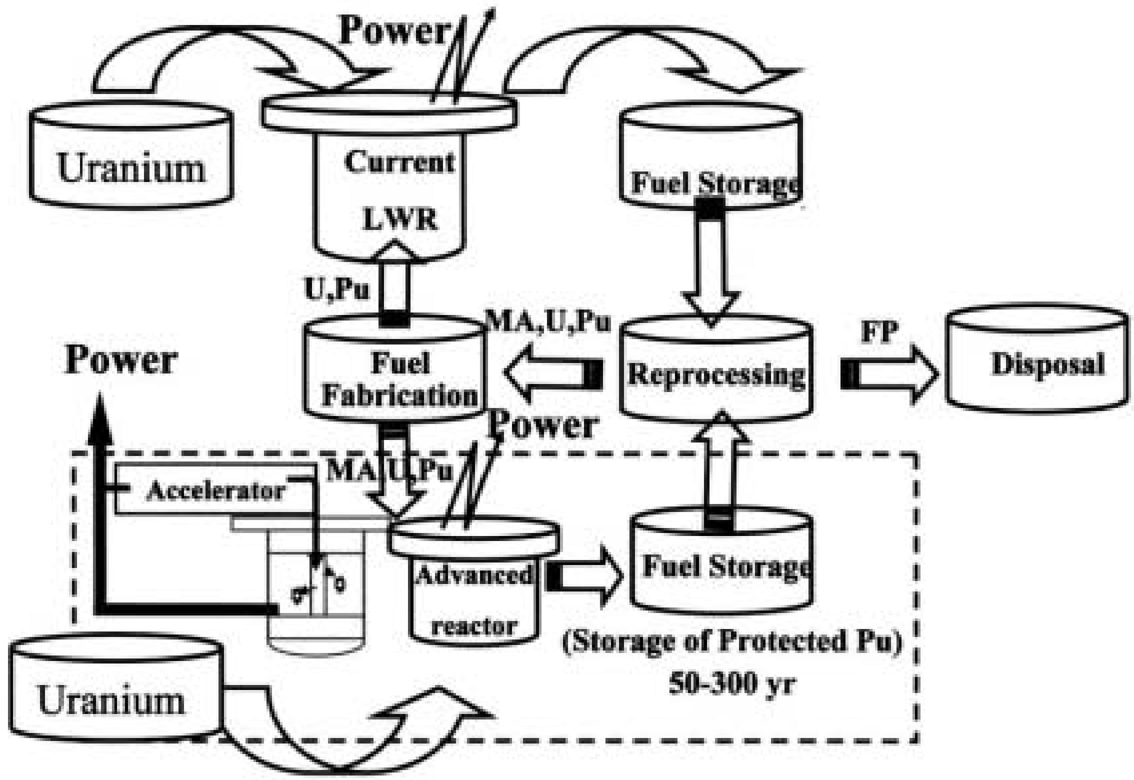


FIG. 14. Proposed advanced fuel cycle with production of proliferation resistant plutonium.

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NEW GENERATION ELASTIC NEUTRON ABSORBER SYSTEMS FOR NEW AND EXISTING DESIGN NUCLEAR REACTORS
Innovative concept

R. GAZETOV, D. SHEVELEV

IPDF, Intellectual Property Development Fund
Visaginas, Lithuania

Abstract. ERCD, the elastic spiral neutron absorber contol concept is presented notifying the advantages and features for specific and existing reactor designs. The advantages listed also covers the new design reactors and positions this innovative concept for use in new reactor design.

1. INTRODUCTION AND BACKGROUND

In many nuclear reactors' designs the power field is controlled by regulation of concentration of liquid or distribution of solid neutron absorbing material in the reactor core. The below-described Elastic Reactivity Control Device combines the advantages of liquid and solid neutron absorber systems. It ensures the uniform distribution of solid neutron absorbing material in the reactor core during the steady-states and transients of reactor.

The ERCD is an elastic absorber element implemented in form of cylindrical spiral spring with the variable, naturally closed, coil gap. Coil gap is increased or reduced by pulling or releasing the elastic spiral with the actuator. In each position of ERCD element the elastic properties of spiral ensure the uniform distribution of absorber material in the reactor core. Reactivity control is carried out by the adjustment of the coil gap L of ERCD absorber with the e.g. rotational actuator drive (Figure 1).

The uniform absorber distribution provided in all reactor operation modes including transients gives the possibility to reach very high uniformity of power distribution in the whole volume of the reactor core, which allows better conditions for fuel utilization to be achieved.

The implementation of ERCD gives the following advantages:

— Flattening of power field gives higher uniformity of power distribution in the reactor core, which can be achieved even in the very beginning of the reactor campaign.
— Better utilization of fuel provided by possibility of achieving of higher degree fuel burnup.
— High performance emergency shutdown - fast introduction of negative reactivity into the reactor core without the axial distortion of power field by rapid and uniform increase of absorber concentration along the axis.
— High efficiency of reactivity control is ensured by ERCD in the whole range of variation of absorber coil gap (concentration), compared to the low efficiency of the conventional control rod in the positions near to the edges of reactor core with low neutron flux.
— High operational availability of ERCD due to the inherent ability of spiral absorber movement in damaged/sagged CPS tubular guides or channels.

— Possibility of implementation of two independent ERCD controls in Dual-Purpose System, which can be placed in one CPS guide (or channel).

Implementation of the proposed Method facilitates the cost-effective improvement of reactor Safety and improvement of Economic Parameters of NPP operation by reduction of fuel share in electricity production cost at NPP due to the better utilization of nuclear fuel. The generic ERCD concept is presented in Ref. [1].

2. GENERAL CONSIDERATIONS FOR APPLICATION OF ERCD

The areas of application of the ERCD are:

i. Fuel Assemblies for
  - Vessel-type reactors
  - Channel-type reactors

ii. CPS solutions
  - Modifications to the existing designs (safety upgrades, modernizations)
  - New reactor designs

2.1. Reactor safety improvement

— ERCD comprises built-in passive safety mechanism for decrease of absorber coil gap, providing self-insertion in the reactor core in case of other equipment failure, which is ensured by the combination of the gravity force and elastic spiral deformation force.

— The possibility of full "withdrawal" of ERCDs from the reactor core is precluded by ERCD design, so that the minimum required level of reactivity margin is provided.

— Elimination of the conditions for positive reactivity release effect in some channel reactors due to CPS water coolant loss by the essential reduction of cooling water volume in CPS channels in the reactor core.

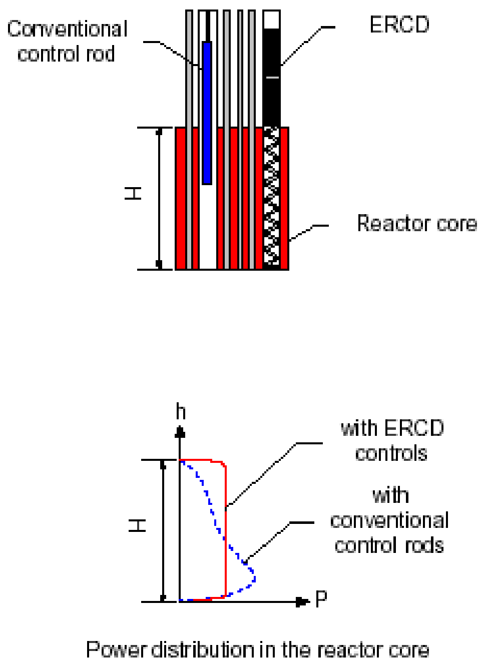

FIG. 1. Uniform absorber concentration effect of ERCD.

2.2. Improvement of economic parameters: Reduction of fuel share in electricity production cost at NPP due to the better utilization of nuclear fuel.

For example, for VVER reactors the additional flattening of radial power distribution of 5-10 % results in increase of average degree of fuel burnup and corresponding reduction of fuel share in electricity production cost at NPP of 1-2 %. The additional flattening of axial power distribution of 5-10% results in increase of average degree of fuel burnup and reduction of fuel share in electricity production cost at NPP of 3-7% [2].

The reduction of fuel cost share in electricity production cost will create better conditions for increase of NPP competitiveness on the energy market.

3. UPGRADES AND MODERNIZATIONS – MODIFICATIONS TO EXISTING REACTOR DESIGNS

3.1. Pressurised Water Reactors (PWR)

The implementation of ERCD in the existing PWR designs is one of the possible solutions that bring the proven technology and existing reactor core design to the next level of economic efficiency and increased passive safety.

The existing solution that is widely used for reactivity control in PWRs is Control Rod Clusters operated either individually (in some designs) or in groups, accompanied with Liquid Poison (Boron) system. The latter provides for the control of reactivity in response to the fuel burnup and ensures the uniformity of the power field that can't be reached with the colntrol rod groups.

Replacement of conventional control rods by ERCD solutions gives more precise reactivity control solution than rods. The axial uniformity of absorber ensured by each ERCD in each moment of time provides the ideal conditions for power field control in the whole reactor core. The required number of ERCD to replace conventional control rods is determined specifically for each design. In case not all control rods are replaced by ERCD, the remaining quantity of control rods shall work as described below.

Key points of ERCD implementation in the Existing PWR Designs are:

i. Basic design of FA remains without changes. Control Rods Drive Mechanism is the same and used for the ERCD placed in the existing control rod guides.
    - ERCD groups are used for power control and compensation of reativity effects - Power control – is implemented by ERCD groups, ensuring high grade uniformity of power distribution during the reactor start-up and the whole campaign. All transients are performed without distortion of power field.
    - Fuel burnup – is compensated by increase of coil gap in ERCD groups.

ii. All other Control Rod groups (if remain in the specific design) are used for compensation of excess reactivity and used only in 2 positions – (1) fully withdrawn or (2) fully inserted (replace the boron control used for this purpose in the original design). During the campaign the CR groups are being withdrawn in stages, as necessary, to keep the ERCD reactivity controls in the required range.

iii. All ERCD and Control Rod groups are used for implementation of Emergency Shutdown.

iv. Liquid Poison (Boron) system is no longer required for reactivity control (can be left for safety reasons as a diverse shutdown system).

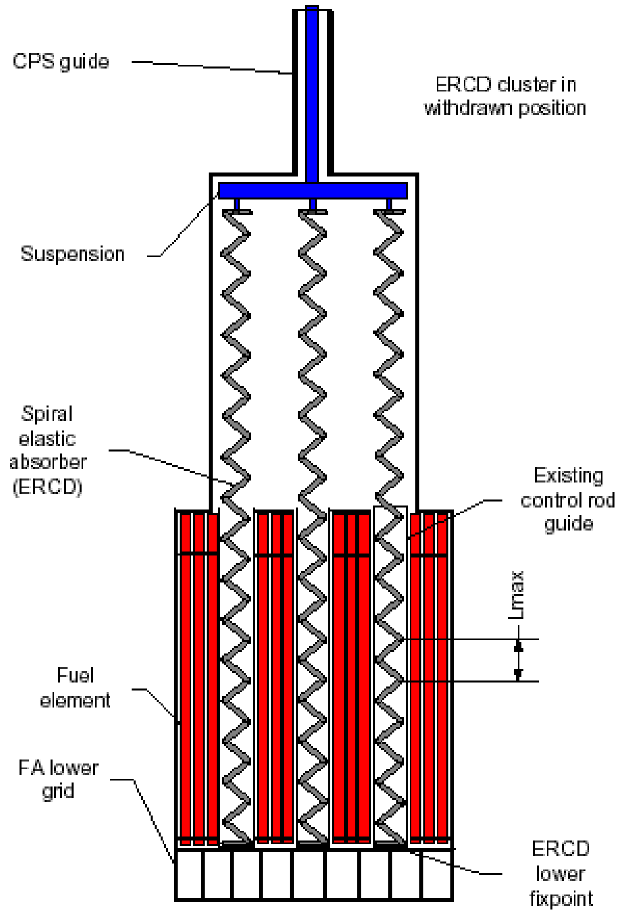

FIG. 2. ERCD in existing design PWR FA.

Results of ERCD implentation:

- Imporved uniformity of power distribution in the reactor core provides conditions for better fuel utilization.
- New safety margins are achievable due to the absense of power field distortion during transients.
- Xenon oscillations are avoided by the method of absorber concentration control.
- Boron-free control concept improves plant economy.
- Minimum scope of modifications to the in existing PWR design.

One of the main problems of implementation (design stage) is the limited ability of the conventional CRDM to actuate ERCD – the range of absorber concentrations achievable will be in range from $C_{max}/2$ to C_{max}. It sets certain limits the selection of absober material and can be addressed by the additional modification to the CRDM to provide wider range of control of absorber concentration.

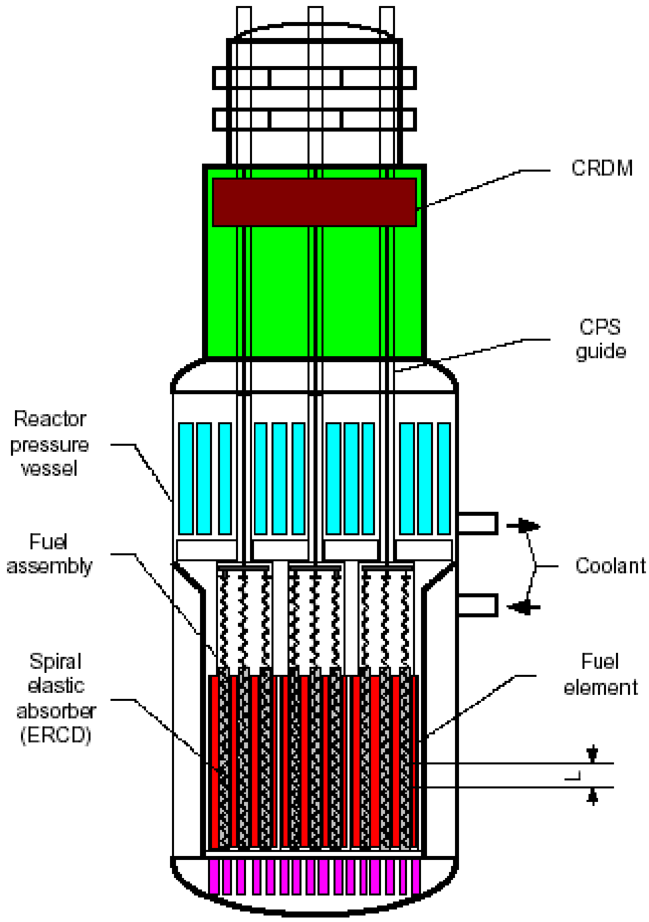

FIG. 3. ERCD in existing design PWR.

3.2. Channel-type reactors

In channel type reactors the reactivity control is performed by Control Rods operating in separate CPS channels that penetrate through the reactor core. Such channels can be cooled by CPS coolant flow. If the Liquid Poison is not used during the normal reactor operation, all reactivity control demands are fulfilled by Control Rods. They are operated individually to reach the necessary precision in reactivity control and power distribution in the reactor core.

Replacement of control rods by ERCD gives all the advantages listed in Paragraph 1, plus provides for the reduction the amount of CPS coolant in the channels in case of water coolant. The latter will eliminate the conditions for positive reactivity release effect in case of loss of CPS water coolant in case of accident.

In most implementations all Control Rods (excludiong those used only for Emergency Protection (EP) function) shall be replaced by ERCD.

Key points of ERCD implementation in the Existing channel-type reactor designs are:

i. Basic design of CPS components in the reactor core can remain unchanged – this include CPS guides and Control Rods Drive Mechanism, which can be left from the original design and used for the ERCD placed in the existing CPS channels.
ii. ERCDs are used for power control, compensation of reativity effects and fast shutdown:

— Power control – is implemented by ERCD, ensuring high grade uniformity of power distribution during the reactor start-up and operation. All transients are performed without distortion of power field.
— Fuel burnup – is compensated by increase of ERCD coil gaps. ERCD controls covers the whole necessary range of control.
— Fast shutdown – is implemented by ERCD and is faster than with the conventional Control Rods due to the presence of elastic deformation force, which ensures the fast insertion of absorber in the reactor core in addition to gravity force.

iii. Emergency Protectioon Rods (if present in the specific design) are used for Emergency shutdown, and provide for the diversity comparing to the rest pool of ERCD.
iv. All ERCD and EP Control Rods are used for implementation of Emergency Shutdown. Two independent systems with the certain extent of diversity and passive emergency actuation are achievable.

Results of ERCD implentation:

(i) Imporved uniformity of power distribution in the reactor core provides conditions for better fuel utilization.
(ii) New safety margins are achievable due to the absense of power field distortion during transients.
(iii) Xenon oscillations are avoided by the method of absorber concentration control.
(iv) Minimum scope of modifications to the in existing channel-type reactor design.

One of the main problems of implementation (design stage) is the limited ability of the conventional CRDM to actuate ERCD – the range of absorber concentrations C achievable will be in range from $C_{max}/2$ to C_{max}. It sets certain limits the selection of absober material and can be addressed by the additional modification to the CRDM to provide wider range of control of absorber concentration In case the replacement of CRDM is included in the scope of modification, the conventional Control Rod actuators and drives are replaced with the rotational actuator drives, which gives the extended range of absorber concentration variation (Figure 4). In case the specific channel reacotr design contains the number of CPS channels insufficient to implement 2 shutdown systems, some of the CPS channels with ERCD may be used for placement of additional independent Emergency Protection rods (dual purpose channel) (Fig. 4).

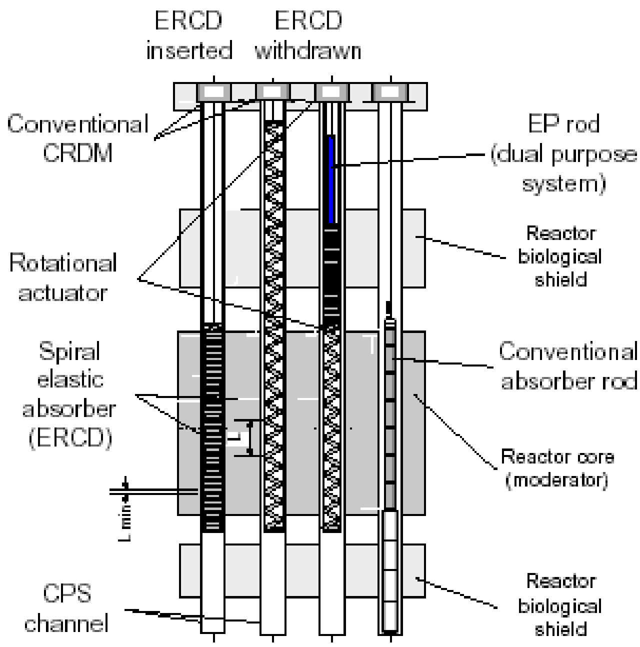

FIG. 4. ERCD in existing design channel-type reactor.

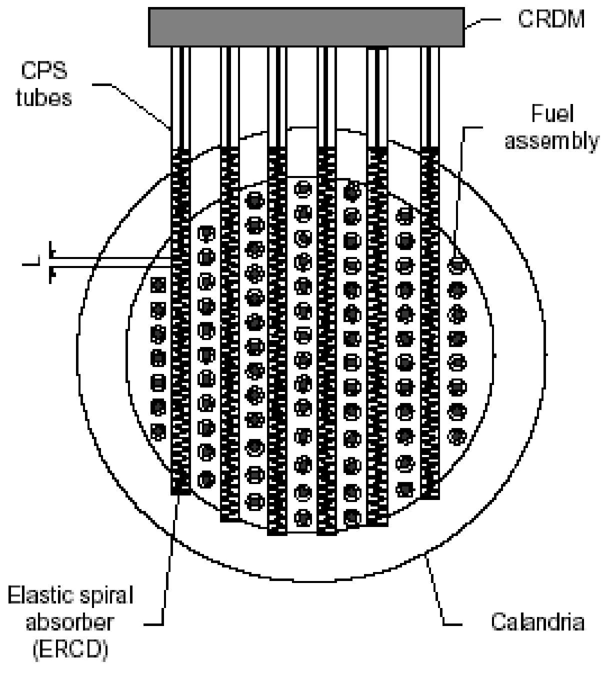

FIG. 5. ERCD in existing design PHWR CANDU.

4. DEVELOPMENT OF NEW REACTOR DESIGNS

The role of ERCD its developers see in the new reactors is primarily to solve the persistent problem of non-uniform distribution of neutron absorber in the reactor core that causes the non-uniform power distribution and all the consequent problems.

Safety-related features of ERCD that will be appreciated for new reactor design are:

(1) Any change of neutron absorber concentration in the reactor core occurs with the high degree of uniformity of axial absorber distribution, including transients. This prevents the distortions and enables guaranteed fulfillment of safety limits.
(2) High speed of insertion of negative reactivity in the reactor core in normal shutdown and Emergency Protection modes, since ERCD absorber insertion.mechanism uses elastic deformation force in addition to “conventional” gravity force.
(3) Passive, “safe failure” concept of ERCD meets the criteria for shutdown systems.

In the new reactor designs all the advanteges of ERCD (listed in Paragraph 1) can be implemented to full extent.

Further specific advantages for New design PWR include:

— Penetrations in the reactor vessel head can be avoided with the use of ECRD with alternative CRDM (possible solution shown in Fig. 6,7)
— Reactor vertical dimensions can be significantly (up to 7 m) reduced in case of use of alternative CRDM
— Use of Liquid poison (boron) for reactivity control can be avoided Additional advantages for new design and modifications of channel-type reactors include the possibility of implementation of Dual-Purpose channel to provide the second diverse shutdown measure in form of absorber rod installed in the free space inside of CPS channel that contain ERCD (Fig. 4).

5. OBJECTIVES FOR RESEARCH AND DEVELOPMENT

ERCD development envisages the following steps to ?? undertake?:

— Neutron-physical calculation of ERCD in reactor core for specific reactor designs
— Mechanical calculations and simulations to determine the set of requirements for structure and absorbing materials
— Determine the implementation options with the regard to materials available
— Develop technical solutions of ERCD for selected reactor designs
— Prototype testing

Specific tasks to perfrom at in the nearest timeframe (years 2003-2004) are

— Neutronic calculations for PWR cores
— Neutronic calculations for channel type reactors on the example of RBMK (large scale graphite-moderted channel type reactor)
— Preliminary set of requirements for selection of material

The completion of these tasks will create a necessary basis to join to international R&D programs and have a number of solutions to contribute to large-scale new designs.

6. ABBREVIATIONS

CPS - Control and Protection System

CR - Control Rod

CRDM - Control Rod Drive Mechanism

EP - Emergency Protection

ERCD - Elastic Reactivity Control Device

FA - Fuel Assembly

PWR - Pressurised Water Reactor

PHWR - Pressurized Heavy Water Reactor

RBMK - Russian design channel type graphite-moderated watercooled reactor

VVER - Russian design pressurized water reactor

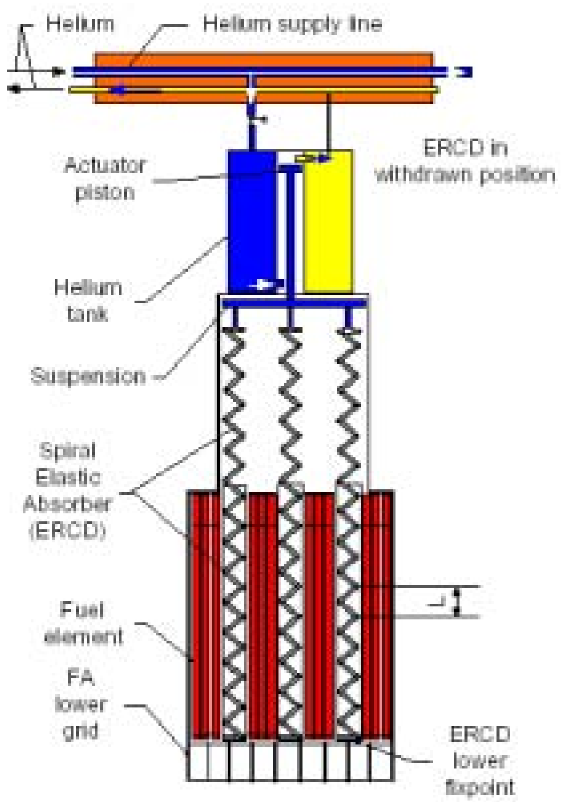

FIG. 6. ERCD in new design FA.

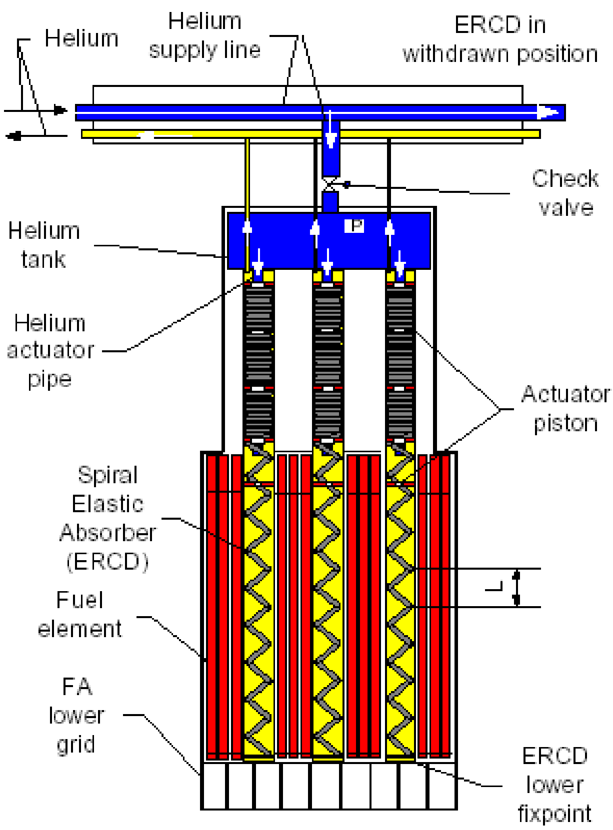

FIG. 7. ERCD in new design FA (with individual actuators).

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INNOVATIVE TECHNOLOGIES FOR THE FUTURE OF NUCLEAR ENERGY IN ROMANIA

I. DUMITRACHE

Institute for Nuclear Research Pitesti (ICN)
Subsidiary of the Romanian Authority for Nuclear Activities
Pitesti, Romania

Abstract. According to the Romanian Nuclear Strategy, all the five units of the Cernavoda NPP will be commissioned. Advanced reactor design is expected for the Units 4 and 5, to be commissioned in the next decade. There is a need of innovative progress. The preliminary evaluation of the already identified technical solutions is presented. Some findings are promising, in particular those related to the performances (based on innovations) of the advanced ACR-700 reactor. However, it seems that a positive coolant void reactivity at the CANDU-6 reactors is related to a significant burnup penalty, for the advanced fuel bundle solutions proposed up to now. No power upgrade is expected in the next decade at the Cernavoda NPP.

1. INTRODUCTION

In the ‘80s, five containments were built on the Cernavoda NPP site, in agreement with a much too ambitious and unrealistic nuclear program. Unfortunately, only one unit, CANDU-6 type, could be completed and commissioned. The commercial operation started in 1996, December. Up to now, the operation was very successful. Work at Unit 2 restarted in 2001.

The current needs of electricity consumption in Romania could be satisfied only by operating many very old and less effective coal power plants. The average cost of electricity generated by them is about 50% higher than the cost of energy produced by Cernavoda NPP Unit 1. Expensive solutions were applied to reduce the SOx and NOx emissions.

The completion of the Cernavoda NPP is related to a prudent investment policy in the electricity generation sector. The schedule for the commissioning of the second unit is dependent on the final decision of co-operation with foreign partners from Canada and Italy. In addition, the Government is ready to negotiate the offers to complete the Unit 3, Cernavoda NPP, under a BOT (Building, Operating and Transfer) or BOOT (Building, Own-Operating and Transfer) solutions. Other types of investment proposals are also accepted for evaluation.

2. NATIONAL NUCLEAR STRATEGY

The Government decided to assume a *National Nuclear Strategy* for the next decades, mainly for the nuclear electricity production and related activities. The non-power applications of nuclear energy are also covered.

The *Strategy* is based on *Objectives*, associated *Specific Strategies* and an *Action Plan*. The practical means to verify and update the *Strategy* are included. There are three distinct groups of objectives:

— The Fundamental Objective,

— The Derivated Objectives and
— The European Integration related Objectives.

Sustainable development criterion played a significant role during the evaluation of the future energy solutions.

The Fundamental Objective specifies that in the period of 2025-2050 years, the production of NPP will cover between 20% and 40% of the total electricity generated in Romania.

The upper limit of 40% is related to the decision to use three significant solutions for power generation, based on: fossil fuel, nuclear and hydro power plants. (The limits of the hydro-electric potential are known. The expected help from the new sources are not very high, due to the characteristics of Romania.) The lower limit of 20% was based on the evaluation of the future need of electricity, on the age of many old and less effective coal power plants and on the natural resource estimation. However, the strategy is very flexible, the future power competition being open for any solution. Although the completion of all the five Cernavoda NPP units is decided, there is no fixed schedule and no selected types for the future reactors.

3. MAIN NEEDS OF INNOVATIVE SOLUTIONS

N1. Lower investment cost and shorter time for commercial operation. The investment cost for CANDU-6 nuclear units, Cernavoda NPP, is relatively high, in comparison against the one for the natural gas based new power plants. Although the containments are already built, the needed time for commercial operation is still long.

N2. Competitive levelized cost for 1 MW·h of generated electricity. Even if the cost of the energy produced by the Unit 1 Cernavoda NPP is very competitive now, a lower one is needed for the next decades, according to the cost predictions for the electricity generated by the future performant gas based power plants.

N3. Much lower mass of discharged fuel for 1 MW·h of generated electricity. The burnup value of the discharged fuel for CANDU-NU (natural uranium) fuel cycle is very low, relative to the current value for the light water reactors. Although the technical requirements for the final HLW repository for low burnup CANDU-NU fuel are less demanding, the ratio of mass of HLW per generated unit energy is too large and, consequently, better solutions are welcome.

N4. Advanced safety features, at least for the Units 4 and 5, Cernavoda NPP. According to the current Action Plan, these units will not be commissioned during the next ten years. Taking into account the permanent public demand for higher safety standards and the already proposed technical solutions for various power reactor types, there is a clear need for an advanced reactor with performant safety features.

N5. Power upgrade. There are significant reasons to assume that the first three units, CANDU-6 type, will be operated for many years, till the middle of the century. For several PWR, BWR and VVER reactors, upgrading of power was already performed with significant success. Innovative solutions are needed for power upgrade at the first three units of Cernavoda NPP.

N6. Improved safety for the Cernavoda NPP CANDU-6 reactors. Almost all of the CANDU-6 safety features were world wide recognized as performant. However, in the next decades, more demanding standards are expected for the reactor safety related characteristics.

4. PRELIMINARY EVALUATION OF THE IDENTIFIED INNOVATIVE SOLUTIONS

4.1. Advanced reactors

Several advanced reactor projects and concepts were analyzed. From the point of view of "competition" for Cernavoda NPP Units 4 and 5 advanced PHWR, PWR and VVER were identified for future analyses. The reactor power must be at least 600 MW. The preferred upper limit is 750 MW, but higher power solutions may be analyzed, if significant advantages are expected. In addition, there are promising advanced reactor projects, other than CANDU, PWR and VVER. They are "compatible" with other NPP than Cernavoda. However, the completion of current work is a difficult task for the next decade and, consequently, no investigation was dedicated for other NPP.

Romanian industry can supply the needed CANDU fuel and heavy water. The high quality of these products was confirmed after more than six years of successful operation of Unit 1. Hence, the evolution of the AECL advanced reactor concepts and projects was analyzed in more detail (www.acr-700.com, [1], [2], [3], [4]). The preliminary evaluation indicates that the Advanced CANDU Reactor ACR-700 may be "adapted" to the already built CANDU-6 containments. According to the available technical and commercial data, the above-mentioned needs for safety, investment cost, construction and commissioning time etc. are satisfied. The investment cost for the ACR-700 is significantly lower than the one for the CANDU-6 reactor. The average burnup of the discharged fuel bundles is about three times higher than the one for the NU fuel, CANDU-6 core. The CVR (coolant void reactivity) is negative. The Romanian plant ROMAG may supply the required heavy water. The FCN Pitesti may manufacture the needed fuel bundles. The innovative ideas applied in the ACR-700 design are presented in the above-mentioned references.

According to the current strategy, the competition for Cernavoda NPP Units 4 and 5 is open for all the advanced reactors of PHWR, PWR and VVER types.

4.2. Significant increase of fuel burnup, CANDU-6 reactor

Early investigations of the ICN Pitesti indicated that a moderate increase of uranium enrichment, to 0.9%U235, for instance, is very promising. The mass of the discharged fuel for 1 MW·h of generated electricity is reduced by more than 40%, the 37-element fuel bundle seems to be enough performant at the associated extended burnup values and the pin power peaking factors remain acceptable. Several years later, these findings were confirmed in [5] and [6]. Unfortunately, the positive coolant void reactivity value for SEU (Slightly Enriched Uranium) fuel was larger than the one for the natural uranium case, by a few percentages. The potential advantages related to the increase of fuel burnup remained unconvincing and the continuation of the investigations was discouraged.

An impressive effort was dedicated to the development of advanced CANDU fuel bundles. AECL developed the CANFLEX fuel bundle, with 43 elements, two distinct pin diameters. The product is already commercial. The proof of performances is based on the results of the needed investigations, including a demonstration test at Point Lepreau NPP, [7], [8], [9]. For the core of the ACR-700 reactor, with CANFLEX fuel bundle, enrichment 2%U235, the coolant void reactivity is negative. For the CANDU-6 core, with CANFLEX fuel bundle, the CVR is positive and increases when the enrichment is higher.

In Argentina, the CARA fuel bundle, with 52 fuel rods, is in an advanced stage of development. Once again, there are significant advantages from the point of view of performances at higher burnup and power. The positive CVR value is higher for CARA-NU than for the CANDU-NU 37-elements, and increases when the initial enrichment increases.

In Romania, the investigations started with SEU 37-element fuel bundle, the main purpose being the significant mass reduction of the discharged fuel, without spending of many years and resources for development and for performance demonstration of an advanced fuel bundle. As mentioned above, the balance of economic and safety results was evaluated as unconvincing and the research was discouraged. However, the effort related to the enriched fuel continued. The fuel specialists tried to develop an advanced fuel bundle, called SEU-43, [12], [13]. An independent investigation effort was related to reactor physics and thermal-hydraulics. Several solutions of bundle type were defined and computer simulations were performed, to look for the numerical values of coolant void reactivity, pin power peaking factors, burnup penalties related to the use of burnable poisons etc. The main result was negative: there is no satisfactory solution for a "simple" fuel bundle design; *for the existent CANDU-6 reactors, a negative coolant void reactivity value is obtained only by significant burnup penalties*. Interesting results were expected from two independent innovative solutions, not mentioned in the literature. Due to the complicated fuel bundle features, the simulation effort is based mostly on Monte-Carlo method and the research progress is slow. The preliminary results are promising, as expected. Unfortunately, the value of the associated fuel bundle price increase is uncertain.

4.3. Negative coolant void reactivity value and low void reactivity fuel

All the CANDU reactors commissioned up to now have positive coolant void reactivity. For the CANDU-6 reactor, this unwanted feature was accepted, based on rigorous safety analysis. As mentioned in 1980, in [14], the intrinsic reactor physics characteristics can assure a small enough power peak during transient, for all the postulated LOCA. However, any improvement of the safety performance is welcome. The negative coolant void reactivity seemed to be possible for an advanced fuel bundle design. The representative investigation results may be found, for instance, in [15] and [11] for the CANFLEX and, respectively, CARA advanced fuel bundles. The associated burnup penalty seems to be unacceptable in both cases.

Consequently, the effort was dedicated for a less demanding task: decreasing of the absolute value of the negative CVR, with a low enough burnup penalty. The levels of the negative CVR and burnup penalty values are dependent on the core features, the CANDU-6 reactor being more stable than the CANDU-9 one.

The positive CVR at the ACR-700 reactor is related on both core and fuel bundle reactor physics characteristics. As mentioned above, innovative ideas and research effort are still needed for a better solution for the CANDU-9 and CANDU-6 design.

4.4. Recovered Uranium, DUPIC and Thorium Fuel Cycles

Due to the excellent neutron economy, the CANDU reactors are very flexible regarding the fuel characteristics. Consequently, Recovered Uranium (RU), DUPIC and Thorium Fuel Cycles are compatible with the existing design of CANDU reactors, [8].

The RU fuel cycle offers significant economic benefit. The research at ICN Pitesti was slowed down by the difficulties in obtaining small amounts of RU. The investigations related to the coolant void effect reactivity indicate that the enrichment of 0.9%U235 may be too low, if burnable poison is used. Taking into account the additional problems related to the transportation of RU at long distances, it seems that the research effort dedicated to this option will not be encouraged.

The DUPIC fuel cycle solution was evaluated as not interesting for Romania, at least for the coming years. The thorium fuel cycle investigations started in the ‘80s at ICN Pitesti. After several years of research, the decision was to make only re-evaluations of performances, benefit and requirements for Romania. However, in the recent years, the answer was clear even without new investigations: no thorium fuel cycle in Romania, at least in the next two decades.

4.5. Power upgrade

The benefit of reactor power upgrade is impressive, in particular in USA and Finland. The use of an advanced CANDU fuel bundle seems to offer the basis for an upgrade. However, several years are needed to have a practical demonstration of performances for a whole core of advanced fuel bundles. Afterwards, the power upgrade solution will become more convincing.

5. CONCLUSION

Romanian Government decided to adopt a National Nuclear Strategy. According to the associated Action Plan, all the five units of the Cernavoda NPP will be commissioned. The operation of the first unit, commercial in December 2002, is successful. The work to complete the second unit restarted. The Government is ready to negotiate the completion of work at the Unit 3. The schedule for the Units 3, 4 and 5 is very flexible. However, it seems that the third unit will be more or less similar with the Quinshan CANDU-6 reactor. Advanced reactor design is expected for the Units 4 and 5, to be commissioned in the next decade.

There is a need of innovative solutions for both the existing and the future reactors. Competitive levelized cost for 1 MW·h of generated electricity, much lower mass of discharged fuel for 1 MW·h of generated electricity and improved safety features are common requirements for all the five reactors of Cernavoda NPP. Lower investment cost, shorter time for commercial operation and advanced safety characteristics are additional needs for the Units 4 and 5. Solutions for power upgrade are needed for the first three units, CANDU-6 reactor type.

Several innovative solutions were already identified. The results of a preliminary evaluation are available. Most of them are promising, in particular the investment cost, the levelized cost for 1 MW·h of generated electricity and the reactor physics innovations for the advanced ACR-700 reactor. Other findings are less satisfactory than expected. Both the CANFLEX and CARA advanced fuel bundle require significant amount of burnable poison (with high burnup penalty) if a positive coolant void reactivity value for the CANDU-6 core is required. It seems that no power upgrade will be in the next decade at Cernavoda NPP.

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THE INTERNATIONAL SCIENCE AND TECHNOLOGY CENTER (ISTC) – TEN-YEAR EXPERIENCE IN SUPPORTING INNOVATIVE NUCLEAR AND OTHER PROGRAMS (INFORMATION REVIEW)

L.V. TOCHENY

ISTC - International Science and Technology Center,
Moscow, Russian Federation

Abstract. The ISTC is a unique international organisation created ten years ago by Russia, USA, EU and Japan in Moscow. Numerous science and technology projects are realised with the ISTC support in different areas, from biotechnologies and environmental problems to all aspects of nuclear studies, including those focused on the development of effective innovative concepts and technologies in the nuclear field, in general, and for improvement of nuclear safety, in particular. The presentation addresses some technical results of the ISTC projects as well as methods and approaches employed by the ISTC to foster close international collaboration and manage projects towards fruitful results.

1. INTRODUCTION

It was more than ten years ago, when the intergovernmental decision has been taken on founding of the ISTC – International Science and Technology Center in Moscow, Russia. It was done in conditions of global political and economical changes in the USSR countries, trends and efforts toward suppression of confrontation of the Great Powers, decrease of risk and store of all kinds of mass destruction weapons, and toward non-proliferation of weapons and weapon technologies.

2. ISTC –HISTORY, STATE-OF-ART, POTENTIAL AND PERSPECTIVES

The basic idea behind establishing the ISTC was to support non-proliferation of the mass destruction weapons technologies by re-directing former Soviet weapons scientists to peaceful research thus preventing the drain of dangerous knowledge and expertise from Russia and other CIS countries.

The Agreement on the ISTC creation as an intergovernmental organisation was signed in November 1992 on behalf of the European Union, Japan, Russia, and the United States of America.

It was declared that the ISTC would pursue the following objectives:

— Give weapons experts in the CIS the opportunity to redirect their talents to peaceful activities
— Contribute to the solution of national and international science and technology problems
— Reinforce the transition to market economies
— Support basic and applied research
— Promote integration of CIS scientists into global scientific community

Thus, the ISTC motto is “Nonproliferation through Science Cooperation”.

The ISTC is registered with the Russian Ministry of Foreign Affairs and enjoys the status equal to that of a diplomatic mission. The Center began its operations in Moscow in March 1994. Since then, the initial parties to the ISTC Agreement have been joined by Norway and the Republic of Korea, as well as by Armenia, Belarus, Georgia, Kazakhstan, the Kyrgyz Republic, and Tajikistan.

2.1. Principles of ISTC operations

- The ISTC solicits, approves, finances, and monitors peaceful science and technology research conducted by Russia and CIS weapons scientists
- Projects are carried out at research institutes and facilities located in Russia and other CIS countries
- In the process of project implementation, collaboration with Western scientists and science organizations is highly encouraged

2.2. ISTC science project program

The Science Project Program is the most comprehensive nonproliferation activity conducted by the ISTC. Through this program, the ISTC solicits scientific project proposals from institutes throughout the CIS and provides funding and logistic support to project teams. Project teams receive written concurrence from the host country on whose territory their research will be conducted, and then develop and execute their project with foreign collaborating organizations. Foreign collaborators ensure that the project goals contribute to the state-of-the-art in the field, and results will find applications to real problems in basic and applied research.

At present, the ISTC is looking to introduce a more programmatic approach towards meeting its goals and, continuing to support individual projects, plans to launch large scale international programs. An example of such a program is the establishment of the International Science Laboratory – a long-term cooperative science facility that would enjoy both the expertise of the hosting science staff and the experience of visiting collaborators.

2.3. ISTC partner program

To better contribute to the solution of national and international science and technology problems, and to match Russia and CIS scientific potential and expertise to the needs of the world science, industries and businesses, since 1997 the ISTC has been pursuing its Partner program.

The Partner Program provides opportunities for private industry, scientific institutions, and other governmental or non-governmental organizations to fund research at CIS institutions via the ISTC. Partners benefit from the ISTC infrastructure and status which permit consistent project management, tax-free direct payments to CIS project teams and duty-free import of project equipment. CIS institutes and project teams benefit from their close cooperation with foreign Partners and the application of their technical skills to important and current scientific and industrial problems.

Summary of Advantages available to ISTC Partners:

- Established ISTC project management infrastructure

— Exemption from all taxes and customs duties on payments and imports
— Direct payments in US$ to project scientists
— Financial control and regular audits, in compliance with GAAP
— Project agreements stipulating rights and privileges of the Partner and Institute
— Host government support and pre-approval for projects
— Strict protection of business confidential information

Presently, the ISTC Partner list includes over 180 organizations and leading industrial companies from all ISTC parties.

2.4. ISTC activities to date

— Over 3860 project proposals registered
— About 30 new proposals received each month
— As of May 2003, about 1760 projects have been approved for funding
— Total funding of the ISTC projects exceeds US $520 million with over US $130 million provided by the ISTC Partners
— Partner contribution to the annual ISTC project funding approaches 50 percent
— More than 600 institutions and 52,000 specialists have received grants through the ISTC

2.5. Other ISTC programs

— Seminar Program: the ISTC organizes and conducts seminars toward heightening the awareness of CIS scientific potential, maintaining strong international scientific cooperation between foreign and CIS scientists, linking scientific potential with technology markets, and establishing cooperation with other international organizations and programs. Seminar topics are of broad technical and global interest and support the objectives of the Center and of other international nonproliferation initiatives.
— The Business Management Training Program is conducted to assist ISTC project managers in developing their general business knowledge, presentation skills, and understanding of intellectual property rights. The training complements the technical aspects of the ISTC project, toward helping the project manager in future commercialization of the project results and in securing funding from sources beyond the ISTC.
— Technologies Database Program: through its contacts with hundreds of research institutes and centers throughout the CIS, the ISTC has uncovered many innovative technical projects either planned or now underway which conform to the nonproliferation objectives of the ISTC. The ISTC established the Technologies Database Program to establish and expand information exchange infrastructure concerning research activities, toward promoting the expertise of CIS research institutes and cooperation between CIS and foreign technical experts.
— The Travel Support program fosters collaboration by reimbursing travel and related expenses for CIS scientists who wish to begin or continue technical consultations on the proposals they submit to the ISTC.
— Communication Support, Patenting Support, and other supporting Programs

3. ISTC PROJECTS ON NUCLEAR REACTOR AND NUCLEAR FUEL CYCLE, AND RELATED ACTIVITY

3.1. Areas of interest

Among two thousand projects submitted to ISTC, there are above one hundred funded and as of yet non-funded projects related to different aspects of NFC - Nuclear Fuel Cycle and Plutonium disposition.

These aspects are:

- General technical and economical analysis of NFC.
- Specific NFCs and branches.
- Reactor as a NFC component.
- New and advanced nuclear fuels and fuel elements.
- Reprocessing of spent fuel.
- Transmutation of minor actinides and fission products.
- Plutonium disposition.
- Fuel transport and storage.
- RAW management and burial.
- Computer and experimental modelling.
- NFC simulators and training centers.
- Nuclear power in space.

3.2. Some examples of ISTC activity in favour of development of new nuclear technologies

3.2.1. Strategy analysis of new nuclear fuel cycle in russia with Plutonium (MOX)

The projects result both methodology, including complex analysis and assessment of economics, environmental safety and none-proliferation aspects, and demonstration of its application for further feasibility and advisability study of weapons and civil plutonium utilization in nuclear power of Russia.

Two principal options of NFC with reprocessing of spent uranium fuel were discussed:

- *Strategy A* - assumes moratorium on the processing of the spent fuel starting in 2005.
- *Strategy B* is based on issuing of license for modification of RT-1 plant for VVER-1000 spent fuel reprocessing.

Comparison of three possible scenarios differed in sequences for stocks of weapons and civil plutonium was accomplished:

- *Successive* weapons and civil plutonium utilization: at first, weapons plutonium is to be utilized until the stock would be empty, then utilization of civil plutonium begins;
- *Parallel* utilization of weapons and civil plutonium implemented concurrently respectively in thermal and fast reactors;
- *Combined* weapons and civil plutonium utilization in mixtures.

3 projects are funded, 10 recipient institutes and 7 Western collaborator institutions are involved in. Budget is near $1.1M.

3.2.2. HTGR – Development of new high temperature gas-cooled reactor and related technology

There is activity mainly in favor to development of Conceptual Design of Modular Helium Reactor with Gas Turbine (GT-MHR) and Plutonium as a Fuel. The projects relate to development of fuel elements (Micro-spheres with Plutonium and multi-layer coating), experimental physics study with the critical stand ASTRA (with heated fuel), design and tests of recuperator, magnetic gears, turbo-compressor and other high-temperature components, including helium seal systems, etc.

13 projects are funded, 4 recipient institutes, 3 foreign collaborators, and partner are involved in. Total budget is near $3.3M.

3.2.3. Development andU use of MOX fuel – physics and technology

This group includes many projects related to different aspects of MOX fabrication and implementation: neutronics analysis, bench-marking and verification through the set of critical experiments (stands in IPPE, VNIITF, etc.), development and update of computer codes and data, advance of fuel production technology, development of new Pu storage and transportation casks, etc.

About 30 projects are funded, over 40 recipient institutes and 74 foreign collaborators, and partner are involved in. Total budget is near $9.3M.

3.2.4. Heavy metal cooled reactor and technology

Principal features of using of lead or lead-bismuth as reactor coolant (neutronic, chemical, thermal, heat/mass transfer processes and hydro-dynamic characteristics) are studied and reviewed through the set of related projects.

Monograph "Natural Safety Fast Neutron Lead Cooled Reactor for Large Scale Nuclear Power" with description of the BREST project basis is prepared for publishing.

3.2.5. Molten fuel reactor concepts

Chemical and thermal behaviors of molten salts are studied with the special loop, constructed in VNIITF. Experimental program agreed with the MOST program of EU and is underway under close collaboration with EU institutions.

Unique concepts of reactor with melt uranium fuel with extreme heat and neutronic characteristics are demonstrated with detailed engineering, experimental and calculation analysis.

3.2.6. Accelerator driven systems

A lot of efforts ware aimed on development of principal components of ADS (target, blanket, accelerator). Basis for this study has been constructed in frames of the #0017 project "Feasibility study of technologies for accelerator based conversion of military plutonium and long-lived radioactive waste".

The unique in the World pilot installation "Spallation Target System TC-1" has been developed, designed, fabricated and tested. Unit includes the proton beam target (1 MWt power), lead-bismuth circuit with pumps and heat-exchanger. The unique Russian experience of use of lead-bismuth eutectic as coolant in nuclear submarines was implemented at this project. The TC-1 System was shipped to USA, for its further testing and study.

Set of integral experiments is supported by ISTC with development and construction of large dimension models of subcritical blanket with neutron source, such as YALINA program (Minsk-Sosny, project #B-070), SAD program (JINR, Dubna, project #2267).

3.2.7. Nuclear fuel cycle with transmutation of RAW

Enormous scope of work has been done for measurement, study and modeling of principal nuclear processes leading to transmutation od RAW (about 20 projects). This program includes measurements of results of cascade-type spallation reaction initiated by high-energy protons or other charged particles, cross-section and product yield after fission and other reactions initiated by neutrons in the energy range up to 30 MeV. Nuclear data for large number of actinide isotopes have been measured, evaluated and processed into files.

In accordance with global trends the ISTC projects develop transmutation technology both for critical reactor environment and for sucritical blankets (accelerator and plasma driven). Results of long-term irradiation of samples with MA- minor actinides - in fast reactors will be presented.

Different fuel materials, including molten salt and molten uranium, are studied to be aimed on transmutation efficiency.

3.2.8. New technology for NPP decommissioning

New concept is developed for the key stages of NPP decommissioning -

Deep study of reactor graphite contamination by radioisotopes after decommissioning of Plutonium production reactors has been done. Content of impurities and radioactivity of graphite blocks and sleeves are measured and systematized for further development of acceptable incineration technology.

3.3. Forms of ISTC activity

The ISTC favours the co-ordination of the project flow through participation at joint project workshops, seminars, topical committees, etc.

3.3.1. The ISTC SAC seminars

The Seminar of Scientific Advisory Committee "New Approaches to Nuclear Fuel Cycle" took place in VNIIEF, Sarov (Arzamas-16), in June 1998, within the ISTC Seminar Program. Seminar program included presentations of leaders of Russian and international programs and ISTC project managers.

3.3.2. Contact experts groups

Several CEGs have been established by ISTC and foreign collaborator institutions, which co-ordinate group of projects related to definite problems, e.g., "MOX and utilisation of Plutonium as reactor fuel", "HTGR-M project", "Transmutation technologies", "Corium Management".

3.3.3. Steering Committees

Within some projects the coordination functions are fulfilled by the Steering Committee, established by Recipient institute, ISTC and foreign collaborators. Regular meetings, workshops and active information exchange help to effective project fulfilment.

SCs have been organised for the projects on “Economical analysis of plutonium fuel cycle in Russia”, “HTGR project concept”, and so on.

3.3.4. Relations with International Organisation

The ISTC maintains close contact with international and national nuclear organisations, such as IAEA, OECD/ NEA, International Nuclear Safety Center (Moscow), International Radiation Safety Center (Moscow), and so on. The goal of this activity is use of available information, concentrated in these institutions, for effective realisation of the projects and for incorporation of its personnel, installations and results into international programs.

4. CONCLUSION

The goals of this presentation are to introduce the ISTC programmes to international nuclear community, and to establish partnership between project recipients, foreign organisations and the ISTC, in order to define areas and forms of possible collaboration in future.

5. REFERENCES

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SOME EXPERIMENTAL JUSTIFICATIONS OF CONSTRUCTIONS OF NUCLEAR REACTORS WITH THE USE OF SOLID COOLANT

I. FEDIK[a], V. DENISKIN[a], V. NALIVAEV[a], U. VISHNEVSKI[b], A. DMITRIEV[b]

[a] “FSUE SRI SIA “Luch"
Podolsk, Russian Federation

[b] “Federal Nuclear and Radiation Safety Authority of Russia”
Moscow, Russian Federation

Abstract. The report justifies an ability of creation and specifies basic advantages of the nuclear power reactor, which is cooled by the solid coolant in the primary circuit. The report formulates the basic requirements to the solid coolant and the conditions of its operation in a nuclear reactor. Five batches of particles of solid coolant with the different characteristics were manufactured during the period of 1994-1999. The report describes experimental facility to research the processes of heat transfer and mass transfer, including stability of flow, heat exchange and wear resistance of the coolant. It gives the heat-transfer coefficient measured at flow of the solid coolant in the round-shaped channel with inner diameter of 10 mm, temperature range of a heating wall from 100^0 C to 800^0 C and flow velocity from 0,10 m\s up to 0,22 m\s in vacuum, argon and helium. The service life tests on a wear resistance of carbon pyrocoated particles have been carried out within 1000 hours. The results obtained allow to carry out estimations of parameters of nuclear power reactors. The report lists parameters of a nuclear reactor of 3750 MW heat power and of 1500 MW electrical power.

1. INTRODUCTION

It is necessary to accomplish at least three tasks, if nuclear reactors are considered to be the primary source of both electric and heating power for the mankind in the future in order to save natural resources (gas, oil) and to reduce gas released into the atmosphere after their burning:

(i) To exclude a probability of the severe accident involving radioactivity release and to reduce radioactivity release into the biosphere if all parts of nuclear energy technologies operate.
(ii) To justify process flowsheets for utilizing fuel inventories which are able to provide economically competitive and wide range operation of nuclear power for a long period of time.
(iii) To exclude a probability of usage of nuclear components to manufacture nuclear weapon.

Modernization of PWRs makes them more complicated and costly. Still a probability of the accident with core meltdown in such reactor is estimated as 10^{-5} 1/year and a probability of radioactivity release beyond the last barrier – containment – as 10^{-7} 1/year.

Assuming that in the future the population of Earth would use electricity from nuclear reactors to support their needs at the level of developed countries, the number of large power capacity nuclear reactors in the world should exceed 10^4.

Therefore, a probability of the severe accident involving radioactivity release would have been $10^{-7} * 10^{4} = 10^{-3}$.

Such probability value is not acceptable for the mankind. It is evident that the accidents involving a release of radioactivity beyond the last barrier should be ruled out.

The modern PWRs feature operating pressure of 16 MPa inside the pressure vessel and the gas-cooled GT-MGR reactor – 7 MPa (a joint project of General Atomics of the United States and Russian organizations). This presupposes huge pressure vessels and piping, which are continuously stressed by internal pressure. Internal pressure provokes ruptures, and any leakage leads to fast loss of coolant.

Uranium resources as nuclear fuel are limited. The development of fast rectors with expanded fuel breeding based on U-Pu cycle is still just experiments.

There is scarce experience of a nuclear fuel breeding in thermal reactors using U^{233} - Th cycle. Higher neutron economy in reactors of any type and with any spectrum allows to increase core – breeding ratio and duration of the fuel campaign and thus to increase essentially a power potential of fuel. The long-lived fuel campaign makes the processing of this fuel inexpedient from technical and economic points of view. The reactor with the use of neutrons allows to solve the problem of thorium entrapment in a nuclear fuel cycle.

The thorium entrapment in U^{233} nuclear fuel cycle of thermal reactors makes in many respects for solving a problem of nuclear non-proliferation, as it is insuperably difficult to process such fuel.

Certainly, as the pure thorium does not contain fissionable materials, so enriched uranium or plutonium mixed with dioxide of thorium should be used to start up such reactors. In the script like this the accumulated weapon nuclear materials can be utilised to start large-scale nuclear power engineering.

Mohamed ElBaradei, the IAEA Director General, in his article *Nuclear Power In Time of Changes* expresses his attitude towards these problems as follows: «In my view, a solution to this dilemma may depend heavily on the development of new, innovative reactors and fuel cycle technologies. To be successful, the new technology must be inherently safe, proliferation resistant and economically competitive.»

We have conducted a work that justifies a possibility of constructing a reactor with untraditional coolant to develop radically new reactors and their cycles with perfect architecture. A solid coolant, for example, the carbon-based one, allows to design the primary circuit of nuclear reactor without excess pressure. Such coolant withstands temperatures up to ~4000°K without a collapse.

The idea of using a solid coolant to cool down a nuclear reactor was considered in the 1960-1970s.

The advantages of such a scheme are too prominent for the experts to fail to notice them. But not a single reactor with such cooling system is known. The patent study has shown that the number of patents on using solid coolant is not very big. The ideas to use a solid coolant in nuclear reactors may be divided into two groups: the use of a fine-dispersed solid coolant and the use of coolant in the form of rather large particles.

The descriptions says that fine-dispersed solid coolant is required to be grinded finely and to have low porosity. The requirements for the coolant of large-size particles were very exact. And many descriptions contain some drawings with explanations. Nevertheless the coolant of large-size particles was excluded from further researches because of the enormously high expenditures. Indeed thermal diffusivity coefficient of solid elements is in principle equal to $\frac{\lambda}{C_p \gamma R^2}$, where λ- thermal conductivity, γ- density and C_p- heat capacity of the material, and R^2 – square of characteristic linear dimension, for instance, sphere radius.

It is demonstrated quite clearly that large-size particles have rather low thermal diffusivity, hence, they require high material consumption to remove heat of the power specified. Because of this deficiency of principle the heat removal systems should have been quite costly even at low power capacity. There are data showing that in case of fine heat transferring particles the heat transfers coefficient of solid surface increases in inverse proportion to the size of particles. Taking the above into consideration it may seem that one could get the best result by making the size of particles as less as possible. However, it leads to a dead-end. It is well-known that very small particles, for example, ultra-dispersed powders can not exist in regular gas environ.

The well-formed surface is highly interactive and tends to create conglomerates of these particles and leads to their adhesion that makes impossible to have a reliable and uniform coolant flow. The thorough patent study has shown that the ideas of only one patent may be used in practice. This is the United Kingdom Patent # 1309883 published in 1973. One of the most important requirements of the patent is a low porosity of the bed and fine grinding of heat transferring particles.

An attempt to simulate the primary circuit of reactor at an experimental facility was done to verify feasibility of these ideas. The facility was built in Tomsk-7 (now the town of Seversk) in 1990. It had a section equipped with electric heaters to simulate the heat release of the reactor core and a cooling section to simulate the steam generator. The flow through the heating and cooling sections should have arranged owing to gravity, and the return of coolant upwards had to be provided for by a worm with electric motor.

Graphite particles of 0.2 to 3 mm obtained by grinding of reactor-grade graphite were used as the coolant. This mixture has a very low porosity.

The pipe diameter, simulating the primary circuit was 80 mm. The attempt to start-up the facility and use it to measure heat transfer coefficients at the heater and cooler was a complete failure.

The graphite coolant did not move either upwards by the worm or downwards by gravity. The second attempt used extremely fine particles of process carbon with size of particles 20 to 50 μm. This substance also failed to move.

Some wording of the patent describing vibrators and pulses to agitate the motion of solid finely dispersed and hollow-free substance became clear.

The analysis of theory and experiments produced requirements to be met by a solid coolant used in the primary circuit of nuclear reactor.

One of the most important requirements is the arrangements for a continuous and homogeneous gravity flow of the coolant through all core sections taking into account the dust caused by wear and some amount of fractured particles. Therefore, the idea is that the mass of particles should resemble a liquid to a certain extend. The particles should be sphere like with average diameter from 0.5 to 2.0 mm and nonsphericity rate not more than 10%. "Angle of repose" of particles to the horizon can be utilised as a validity criterion of particles which should not exceed 25^{o}. The heat transfer coefficient should be increased up to the practical maximum value. Other basic requirements:

— A high mechanical strength, ability to resist to impact load.
— High wear resistance.
— High heat resistance.
— Chemical inertia to structural materials of the reactor.
— Low ability to adsorb and to evolve different gases.
— Stability of structure at a long-period operation.
— High heat conductivity and heat capacity.
— High refractory quality and fire resistance.
— Low speed of sublimation and evaporation.

In 1996 – 1997 the system of experimental facilities were built in the Scientific and Research Institute “Luch” to prove the possibility to reliably cool a nuclear reactor with a flow of solid particles and to obtain a minimum set of data for the conceptual design of such reactor with solid coolant.

In 1994-1999 5 batches of particles of different size were fabricated in accordance to different technologies. Four batches were graphite-based and one was aluminium oxide-based (Al_2O_3). The purpose was to verify how the heat transfer coefficient was changing as the particle size varied. The average diameter of graphite particles was 0.5 to 1.3 mm and they were of different quality. The quality of particles was determined by:

— range of average diameters of particles - $(d_{max} + d_{min}) /2$;
— range of nonsphericity - d_{max} / d_{min};
— surface quality of particles.

Particles from the batch #5 were carbon pyrocoated ones with gas-core settling (Fig.1).

The research of stability of the solid coolant flow was conducted on the special facility simulating the setting of the reactor core. The flow stability was proved by time stability of emptying measurement container. The flow structure was checked by sight by fast-track filming through a transparent wall.

An experimental facility was constructed to research heat transfer of a particle flow. It was the working section of the core simulator which represents a pipe of stainless steel with an inner diameter of 10 mm. It has a heater and container - coolers on both ends. The length of a pipe is about 3 meters. The start of motion under gravity is implemented by turning over the pipe. It lasts till particles are poured from upper container into the lower one. The pipe has electrical heating from its outer side. Thermocouples and thermal isolation are set up from the outer side (Fig. 2). To compensate thermal losses, fixed electrical power was brought to a stationary state in temperature, when all power was spent for losses in air without motion of cooling particles. And then working channel of the facility was turned over on 180^{o} and cooling particles came in motion. By increasing the power of heaters, temperature of a heating wall, which was reduced on transiting a particle flow, returned to its basic value, so that loss of heat in the air was identical. The difference of electrical power was evaluated in a heat transfer coefficient taking into account the temperatures of the heating wall and average temperature on cross-section of particles after transiting the heater.

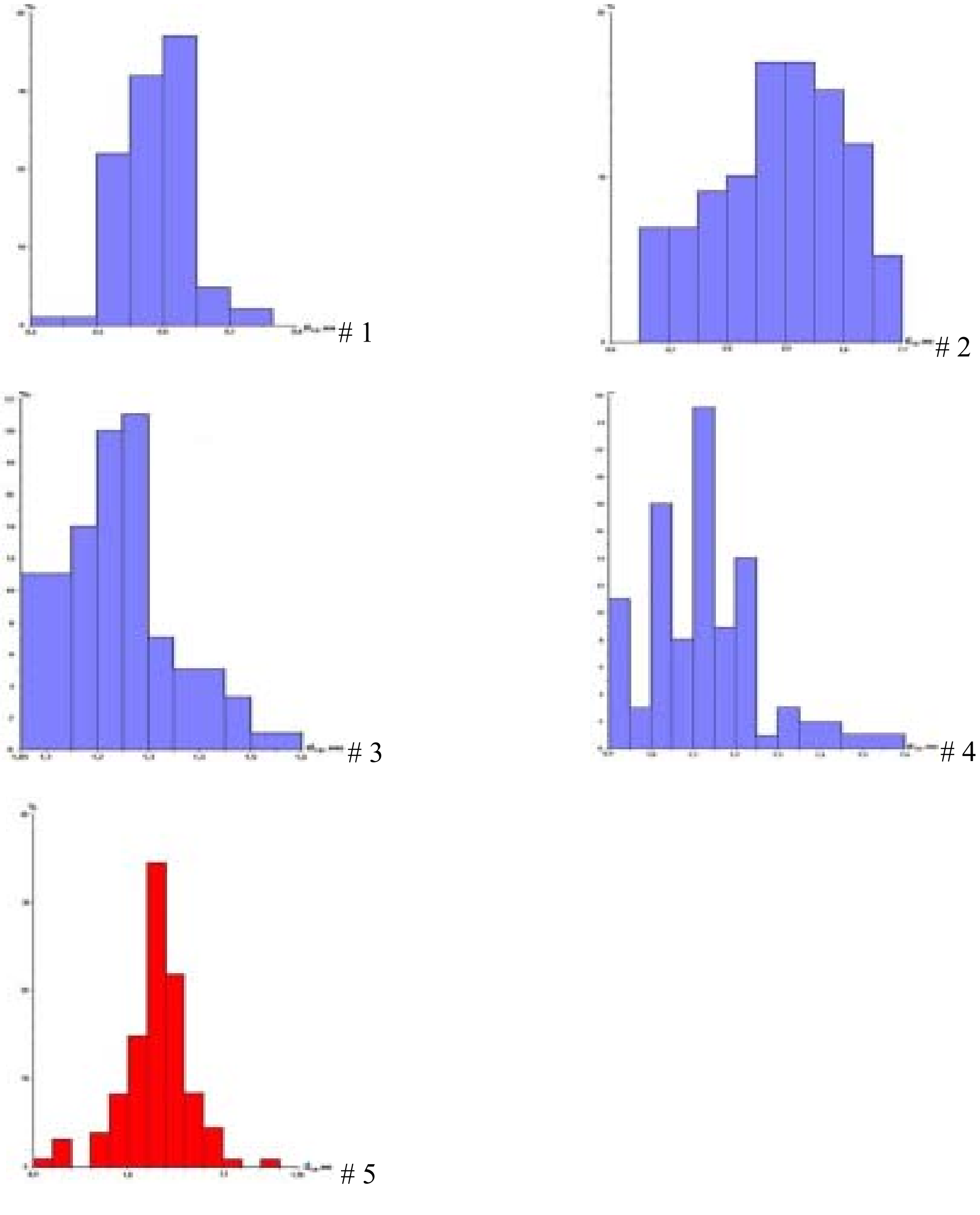


FIG. 1. The histogram of distribution of average diameter of particles for lots # 1-5.

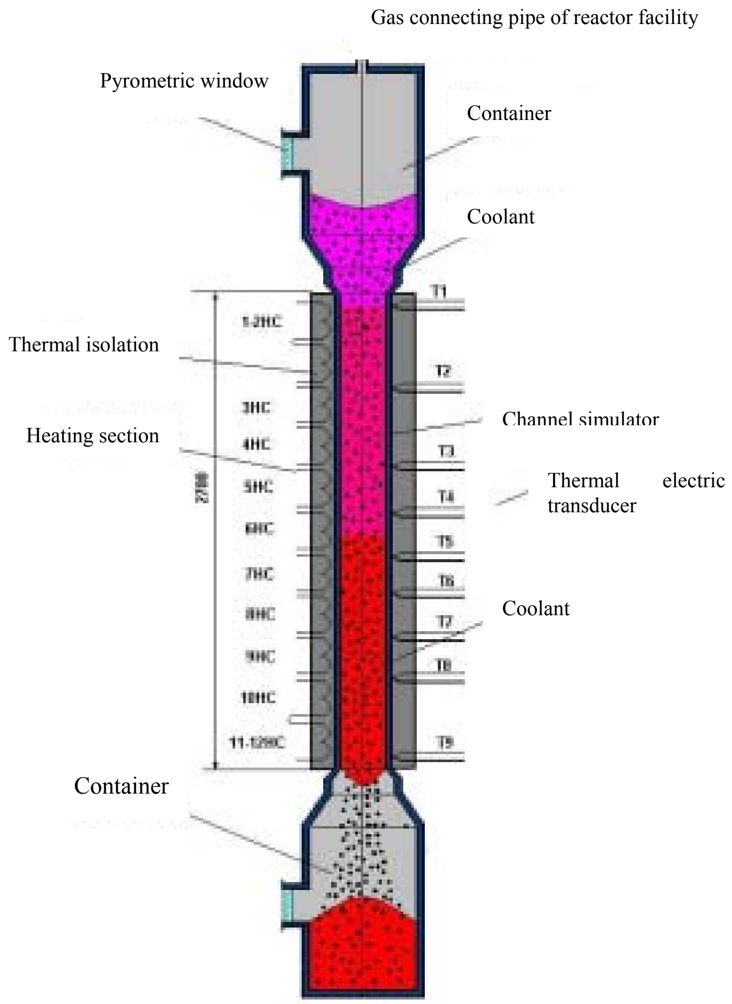


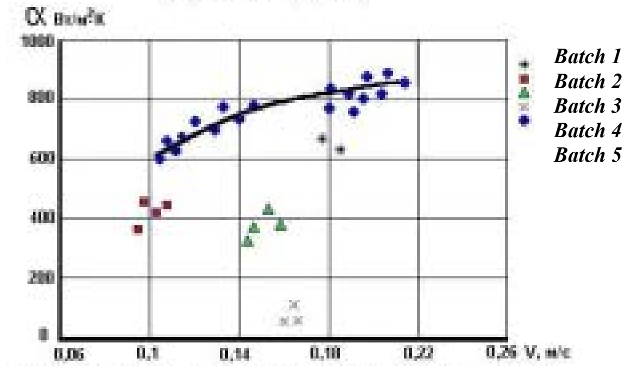
FIG. 2. The facility to determine heat transfer coefficient.

At present we have received the data on heat transfer flow of the solid coolant. The data depend on temperature of a heating wall (up to 800^0C), the velocity of a continuous particle flow (the velocity for our geometry is up to 0.22 m\s), gas environ (helium, argon, vacuum), diameter and degree of particle perfection. The heat transfer coefficient reached the value of 800-1000 W/m^2 K (Fig. 3) in the channel with round cross-section diameter of 10mm without turbulence. The coefficient was reached in the best conditions (helium, velocity of 0.2 m\s, temperature of a wall 800^0 C, a batch of the carbon pyrocoated particles).

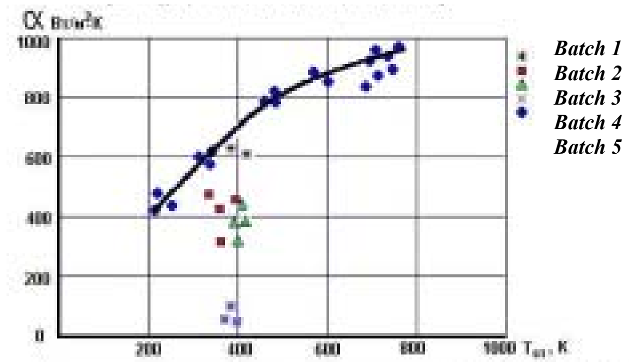
One more facility was generated to measure heat transfer coefficient, on which long-run stationary state of "heating-cooling" was implemented. It was made to make the results of determining heat transfer coefficient more reliable even at high temperatures up to 1000^0 C. The geometry with round channel with outer heater has been used so far.

At the same time the facility allowed to receive the data about wearing of particles after quite large quantity of cycles in the operating mode.

Effect of flow velocity on heat transfer coefficient (T=400±30^{0}C), argon



Effect of temperature level on heat transfer (ΔT=T-T=200-500K, V=0.15-0.19m/s)



Effect of gas on heat transfer coefficient (T=(450±30)^{0}C, V=(0.12…0.8)m/s)

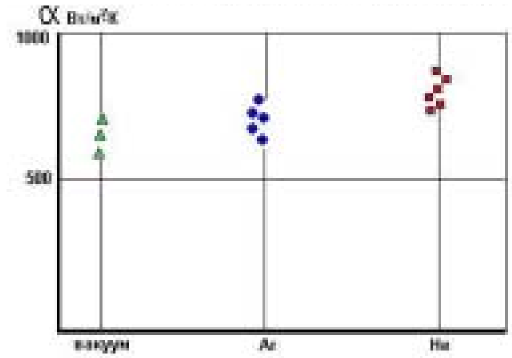


FIG. 3. Dependence of heat transfer coefficient on the different factors.

The facility represents a model of a fuel element of the reactor with the solid coolant (Fig. 4). The working channel of the facility is graphite bushes with overall length of 3 m and inner diameter of 10 mm. There are heaters along the length of about 2.5 meters the power of which can reach 15 KW. Heat exchanger is located below, as well as the device of return of heat transfer particles. The latter returns cooled particles as a chain ribbon with fixed containers to an upper hopper.

Total number of temperature detectors (thermoelectric transducer) is 31 pieces. The facility is equipped with necessary measuring instruments and controls. It has a **pressurized** vessel, which allows to create and to retain necessary atmosphere. The tests of wear resistance of graphite carbon pyrocoated particles were conducted on the facility.

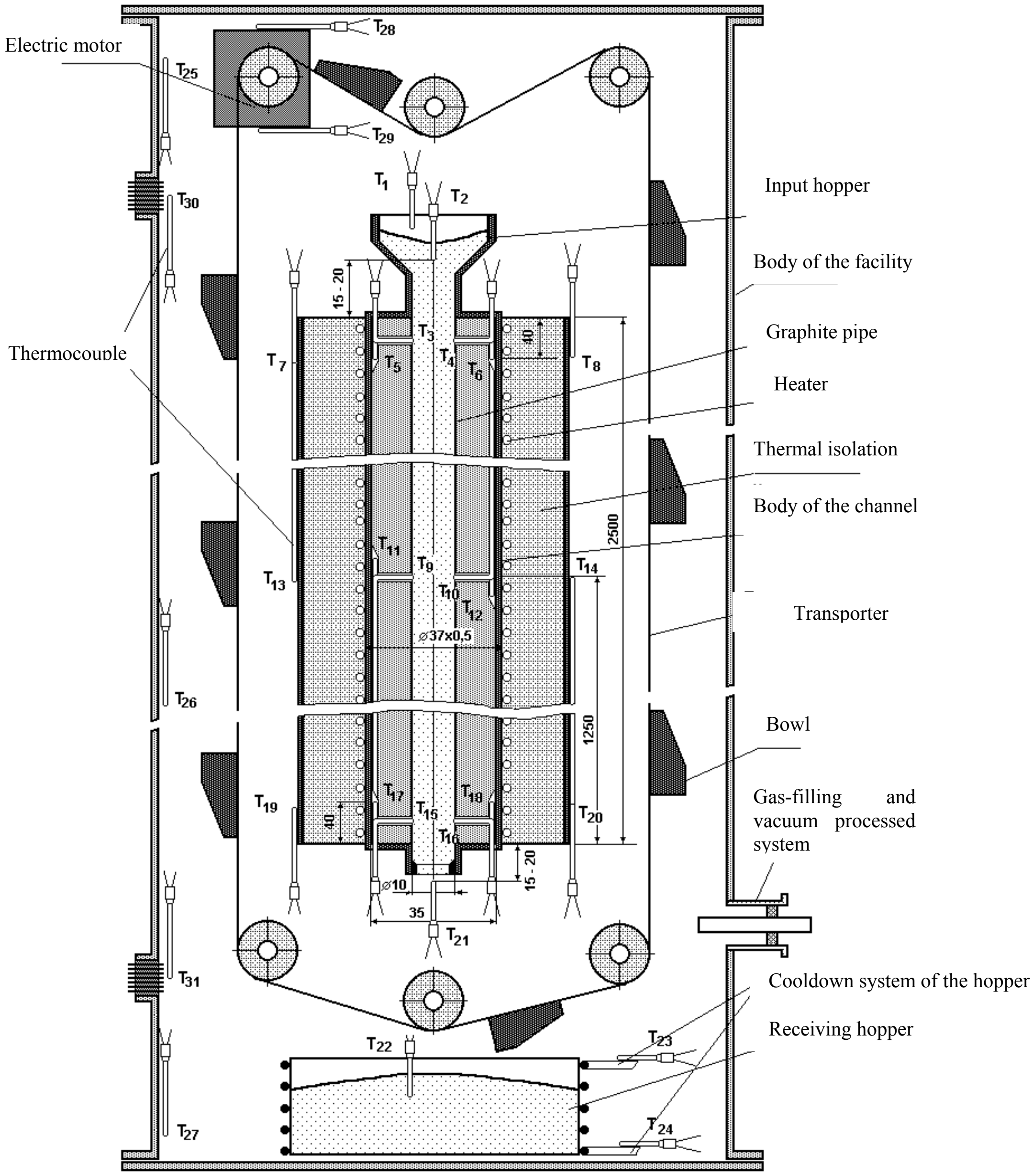


FIG. 4. Model of a fuel element of the reactor with the solid coolant.

The tests of particles of the given batch were conducted during 1000 hours at temperature of 300-400^0 C and with flow velocity of ~ 0,1 m\s and they didn't reveal any destruction. It was recorded only a small change of geometrical average sizes and feeble increase of a surface roughness of particles within the limits of 1-3 microns. This is the result of surface corrosion and mechanical wearing.

Fig. 5 shows the appearance of particles before the tests at magnification of x50, and Fig. 6a shows surface state of particles at magnification of x500 and x1000 (see Fig. 6b)

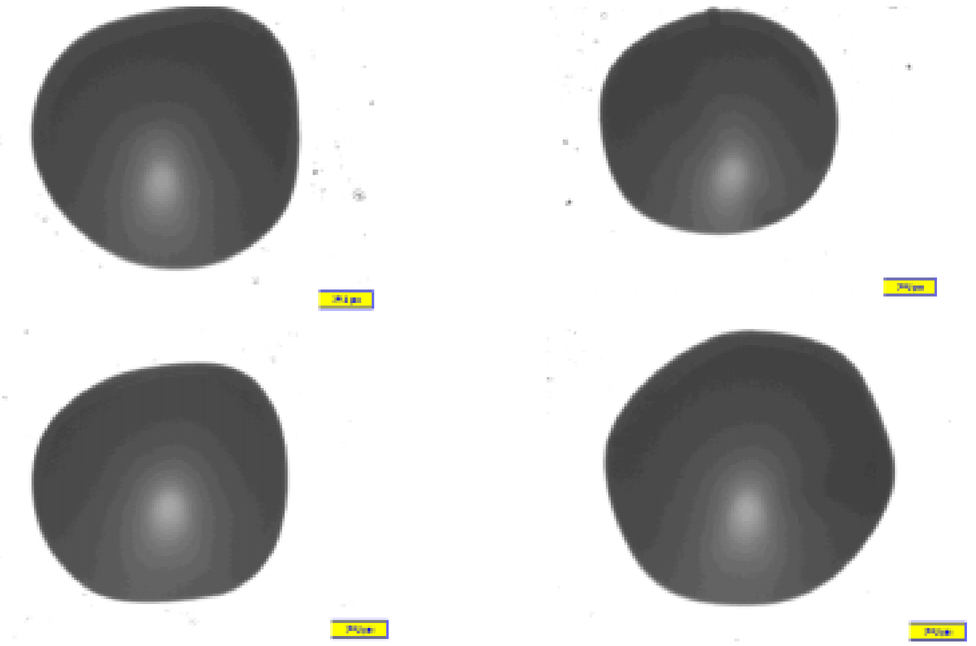
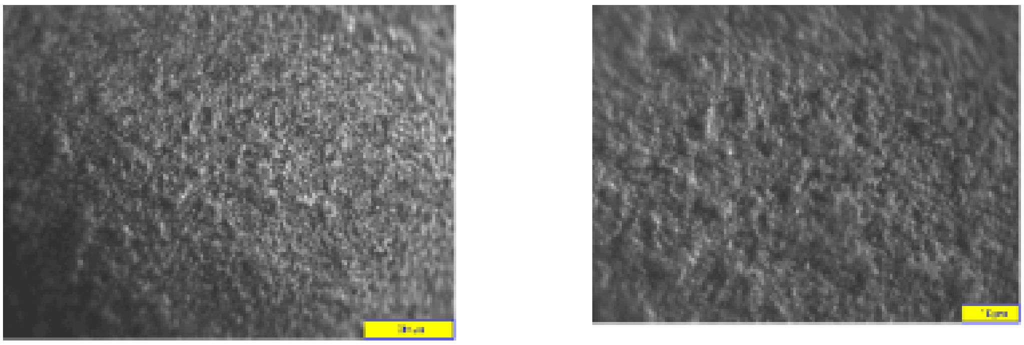


FIG. 5. A general view of particles of the solid coolant before tests, x50.



a) x500; b) x1000;

FIG. 6. The surface of particles of the coolant before tests.

Fig. 7, 8, 9 contain the results of the visible analysis of particle state after 1000 hours of tests.

The average diameter of a graphite core of particles is ~200 microns. The width of carbon pyrocoating changes from ~200 microns up to ~400 microns, taking into account the distributional histogram of average diameter of particles from sampling of 200 particles. The results of the visible analysis of microsections display that the sizes of particles and the state of carbon pyrocoating on cross-section have not been practically changed.

The research of a surface of particles of the solid coolant in an initial condition and after tests within 1000 h was conducted on a scanning electron microscope “Tesla” at magnification of 280-2100. 10 particles in each group were examined.

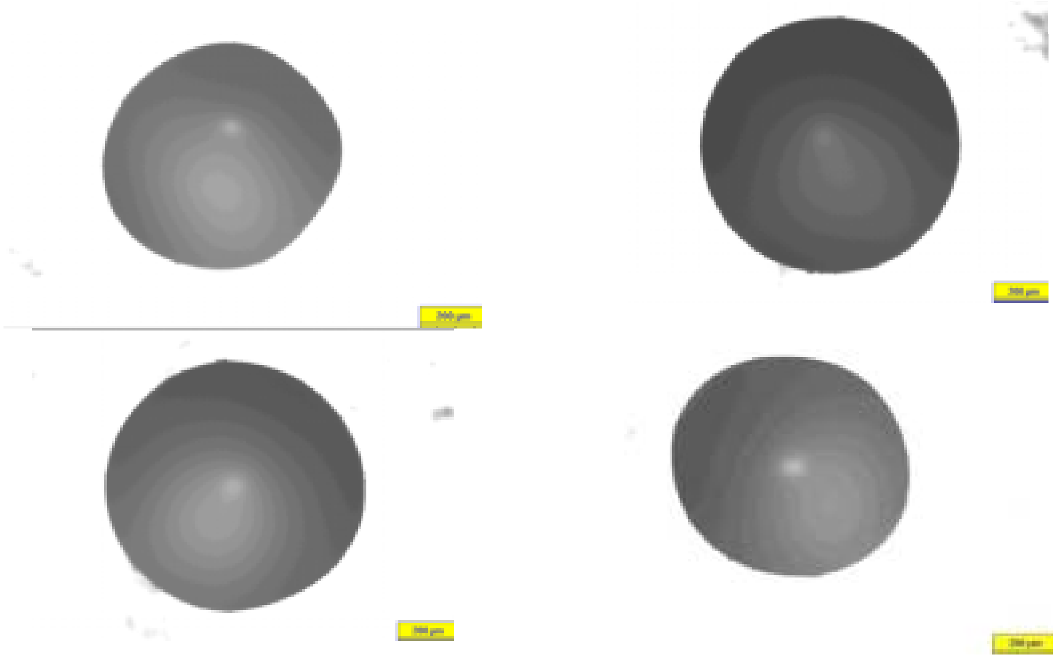
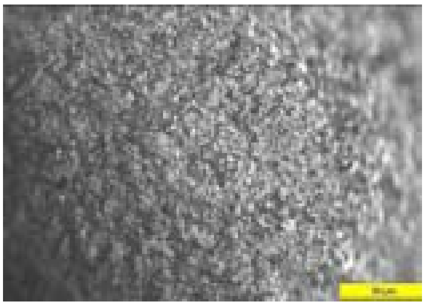
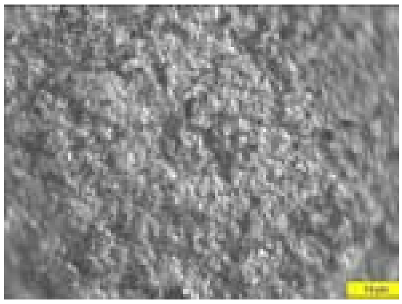


FIG. 7. Appearance of particles of the coolant after tests within 1000 hours, x50.

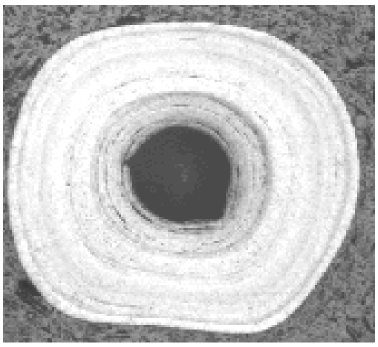


a) x500;



b) x1000;

FIG. 8. The surface of particles of the coolant after tests within 1000 hours.



a) x50;

b) x50;

a) Before tests; b) after tests.

FIG. 9. Microsections of cross section of particles.

a) x280 b) x650 c) x2100

FIG. 10. The view of particles after tests within 1000 hours at different magnification.

The microrelief of the surface of the particles after tests is more clear-cut. The characteristic average size on the surface is ~3 microns.

Figures 11 and 12 display the histograms of particle distribution on average diameter and the histograms of nonsphericity of particles in an initial condition and after tests within 1000 hours.

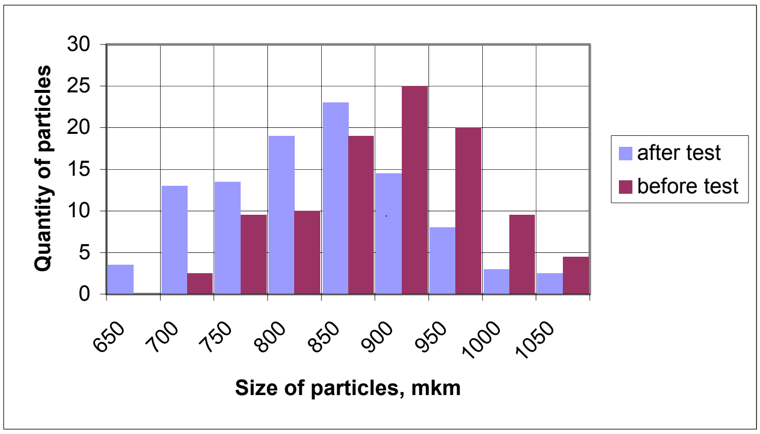


FIG. 11. The histogram of particle distribution on average diameter. (Quantity of sample - 200 pieces).

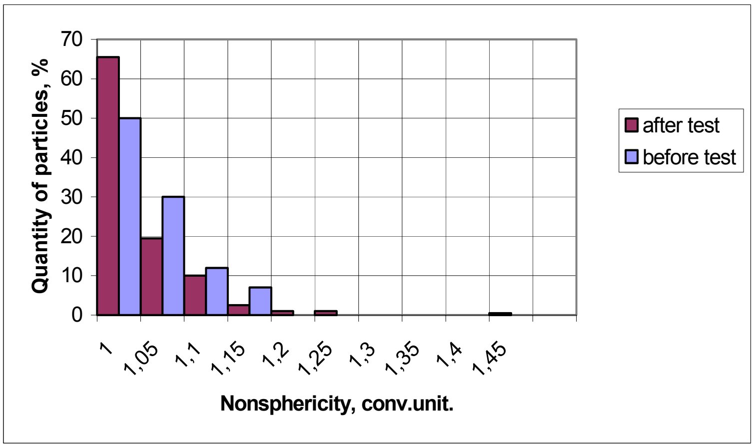


FIG. 12. The histogram of distribution of particle nonsphericity before tests.

The results of the researches formed an invention formula and the patent was issued in Russia in 05.10.2001. The formula and description of the invention give the description of requirements, which allow to conduct the reliable uniform motion of particles with a high heat transfer coefficient, range of diameters of particles, degree of particle nonsphericity, carbon pyrocoating, and also characteristic specific devices necessary for construction of such a reactor.

The analysis of results of experiments displays, that the values of velocities of continuous motion of the coolant and heat transfer coefficient allow to design a nuclear reactor with density of heat release 8-16 MW/m^{3}. The schematic diagram of such a reactor is given in Figure 13. The results obtained were used to conduct the estimations of parameters of nuclear power reactors. The parameters of a possible nuclear reactor are given below.

Fuel coated particles elements are used for the reactor. They are based on U enriched up to 10%, gathered into compacts for GT-MGR type (compact dimensions: 12.5mm diameter, height – 50 mm). Compacts of 10 pieces are placed in vertical graphite shroud pipes of 520 mm length.

The fuel pipes are assembled in the form of square lattice with a pitch of 32 mm. Fuel elements are held by spacers in the fuel assembly which at the same time work as tabulators. The total number of fuel elements in the fuel assembly is 312 pieces. Fuel elements are centred in the core by vertical channels. The outer side of the square is 570 mm.

The core is a vertical cylinder of diameter of 9.5 m, and of 6.5 m height. Total number of fuel assemblies is 2790 pieces. Besides, one row of fuel assembly will be made of depleted uranium or thorium in standard geometry to absorb the neutron leakage around the core radius and lower butt. Axial power picking factor is 1.43, radial power picking factor is ~1.25. The power picking factor of the reactor is taken to be 1.79.

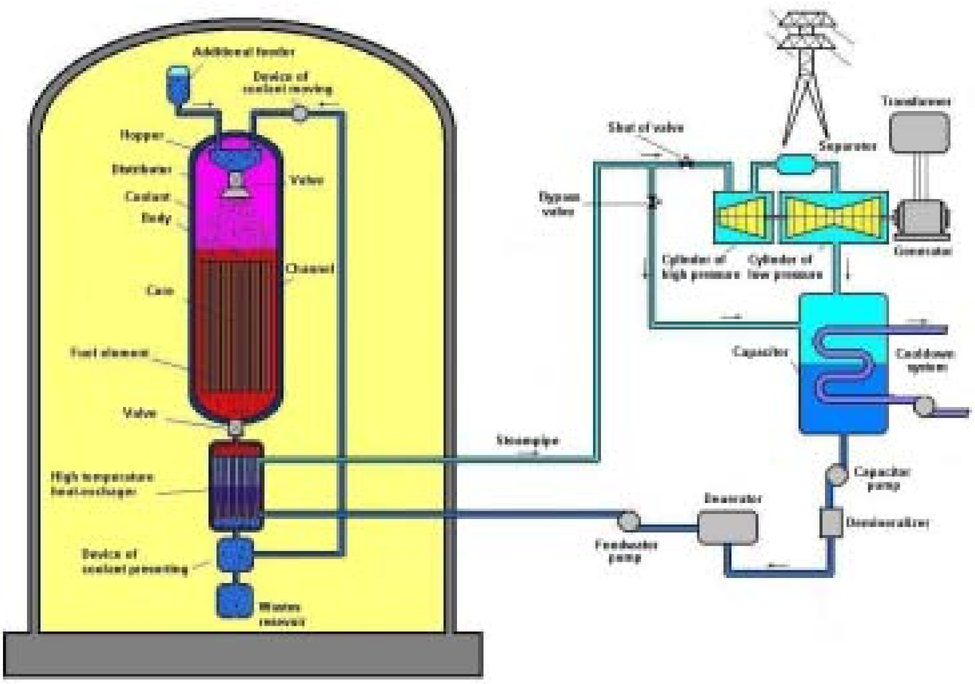


FIG. 13. The schematic diagram of the reactor with the solid coolant.

The volume is filled with helium at a pressure being 0.08 MPa.

The flow rate through the core will be ~5.4 t/s at velocity of the coolant of ~ 0.17 m\s. The reactor power will be ~ 3750 MW at inlet and outlet temperature of ~ 500^o C, ~ 900^o C, respectively. The electric facility power will be ~ 1500 MW at efficiency ~ 40 %. Average thermal power will be ~ 8.65 MW/m^3 in the core, and maximum will be 15.5 MW/m^3.

Certainly, we should conduct large work to justify and to optimise the basic design solutions on such reactor. However, there are no any obstacles for its construction or any reason of its diseconomy.

In conclusion it is necessary to stress the basic advantages of the solid coolant:

(i) Pressure in the primary circuit of the reactor is below the atmospheric one and, as a consequence, there is small steel intensity and cost of the facility. There is a possibility to build a large-power capacity reactor with a low specific power density of the core and high critical margins, which would spare efforts and money to manufacture a complicated and costly equipment and augmented equipment. It means that it is feasible to reduce a possibility of emergency state, to augment safety during all possible accidents, including depressurisation.

(ii) High temperature of the reactor primary circuit allows obtaining a high thermal efficiency coefficient.

(iii) A circumstance of importance is that there are no practically any corrosion related problems while using the solid coolant and the erosion issue may be minimised. In its turn, this means that the system for the coolant treatment and recovery may be simple in design, cheap and cost-efficient in operation.

(iv) The reactor plant may be designed in such a way that its cost and dismantling complexity would be significantly lower than that of existing PWRs.

(v) Radioactive waste generated in the course of dismantling of such a reactor would have a specific radioactivity level and total radioactivity hundreds of times less than that of the existing reactor systems. This does not pose a problem with building a new reactor on the decommissioned site and allows reduction of the number of NPP sites.

(vi) Such reactor practically does not generate liquid waste, and degasifiers may dispose of the minimum amount of gaseous waste generated.

The solid low activity operational waste does not incur large storage costs. The reactor will have good neutron and physical properties.

ENHANCEMENT OF MOX-FUEL INHERENT PROLIFERATION-RESISTANCE *

E.F. KRYUCHKOV, V.B. GLEBOV, V.A. APSE, A.N. SHMELEV

Moscow Engineering Physics Institute (State University)
Moscow, Russian Federation

Abstract. There are analyzed the possible levels of MOX-fuel protection against uncontrolled proliferation within the frames of the closed fuel cycle of nuclear power reactor. Proliferation protection of MOX-fuel at different stages of nuclear fuel cycle is compared with well-known Spent Fuel Standard of typical light-water reactors.

1. INTRODUCTION

Creation of inaccessibility conditions for any unauthorized actions with MOX-fuel is one of the most important components for wide implementation of the closed uranium-plutonium fuel cycle into nuclear power system. So, the measures which are able to give a self-protection property to MOX-fuel represent an important factor in development of protection-in-depth system for MOX-fuel.

One of such protective measures may be the inherent radiation barrier appearing inside of MOX-fuel after neutron irradiation. The inherent radiation barrier of MOX-fuel assembly (MOX-FA) may be characterized by the following main parameters:

— proliferation protection level L in terms of rate of equivalent dose (RED, rem/h; 1 Sv/s = 3.6 10^5 rem/h) at specified distance from MOX-FA;
— time T for continuous action of the inherent radiation barrier at the proliferation protection level above L;
— time-dependent non-uniformity of the proliferation protection level for specified time T of the inherent radiation barrier action.

It is desirable that the proliferation protection of MOX-fuel would be at level that makes any unauthorized actions including short-term actions impossible. In such a case, the MOX-FA proliferation protection must correspond to receiving the lethal radiation dose within a minute range, i.e. RED from MOX-FA must be about 27 krem/h at 30 cm-distance from MOX-FA.

It is accepted that irradiated fuel of power nuclear reactors has the self-protection property due to sufficiently high inherent radiation barrier of long-term action. So, irradiated fuel meets the criterion on effective resistance to short-term unauthorized actions of potential proliferants. In Russia, irradiated fuel is considered as a nuclear material of the lowermost

* This work was performed under financial support of Education Ministry of Russian Federation within the framework of the grant on fundamental researches in technical sciences branch on 2003-2004 years (section 7.4 “Nuclear and Fusion Reactors”).

attractiveness category regardless of its amount [1]. In the USA, the Spent Fuel Standard (SFS) is accepted for characterization of the nuclear material self-protection property, beneath which a risk of unauthorized plutonium diversion is estimated as a significant one [2]. The Spent Fuel Standard is understood here as a time dependence of RED in direct vicinity from irradiated MOX-FA for the first 100 years of its storing.

Fresh nuclear fuel has no the self-protection property, and thus fresh fuel is vulnerable to unauthorized actions. So, profile of fuel radiation protection, i.e. dependence of the radiation protection barrier value at the different stages of fuel cycle, has marked gap at the fresh fuel management stage. In the present paper there are considered those protective measures which are able to form an enhanced protective profile of the fuel during the whole cycle including fresh fuel management. These measures imply introducing small amounts of additional components (different from the traditionally used burnable absorbers, like gadolinium compounds [3]) into fuel composition.

The proliferation protection levels of fresh MOX-fuel containing small additions of protactinium are evaluated for equilibrium closed cycle of a light-water reactor (LWR). It is demonstrated these measures make it possible to close "windows of vulnerability" in the fuel protection profile and provide the self-protection property to fresh fuel at level of the Spent Fuel Standard.

2. PROLIFERATION PROTECTION OF MOX-FUEL BY MEANS OF SMALL PROTACTINIUM ADDITIONS

Presently, eighteen protactinium isotopes are known [4]. Half-lives of all protactinium isotopes are very short, except of ^{231}Pa ($T_{1/2}(^{231}\text{Pa})$ = 32760 years).

Introduction of protactinium into fuel composition leads to the fact that, under neutron irradiation (for example, when MOX-fuel is used in LWR), ^{231}Pa is converted into ^{232}U via the following isotopic chain:

$$^{231}\text{Pa}\ (n, \gamma)\ ^{232}\text{Pa}\ (\beta^-, 1.31\text{ day})\ ^{232}\text{U}.$$

^{232}U takes a particular place amongst the isotopes-emitters of gamma-radiation. High-energy gamma-rays emitted by ^{232}U is caused by its decay products, mainly by ^{208}Tl (see Fig. 1). Thus, as a result of ^{232}U generation, there is provided the long-term protective radiation barrier which, under uranium recycle, is transferred naturally to fresh MOX-fuel.

Artificial isotope ^{231}Pa can be generated, for example, in irradiation of natural thorium by high-energy neutrons (E_n > 6.35 MeV):

$$^{232}\text{Th}\ (n, 2n)\ ^{231}\text{Th} \xrightarrow[\text{25,6 hours}]{\beta^-} {}^{231}\text{Pa}\ .$$

Generally speaking, under natural conditions, isotope ^{231}Pa is permanently generated in α-decays of ^{235}U, and both these isotopes are in equilibrium state with each other. ^{231}Pa is the only natural protactinium isotope accessible in weight quantities. However, content of protactinium in uranium ores is very low, close to content of radium, and it is equal to ~ (3÷4) 10^{-5} weight % in respect to uranium. So, protactinium is a natural element of low abundance.

Protactinium can be easily separated from uranium by simple chemical technologies [4]. It allows to increase significantly protactinium content in uranium products. Undoubtedly, this problem requires a separate study from standpoint of developing an effective, economically competitive technology.

Protactinium content in nuclear fuel can be varied within rather wide range, from very small amounts, allowing to enhance only a traceability of the diverted material, to significant amounts which can lead to receiving the lethal radiation dose for relatively short time of unauthorized fuel handling.

Preliminary evaluations have shown that increase of protactinium content in natural uranium by a factor of 300 (up to several hundredths of percent) can enhance a fuel traceability from radiation background data up to significant RED values (about 100 rem/h at 1 m-distance from FA). If protactinium content in fuel is further increased, then ^{232}U can be accumulated in quantities which are able to give the self-protection property to fresh MOX-fuel at level of the Spent Fuel Standard.

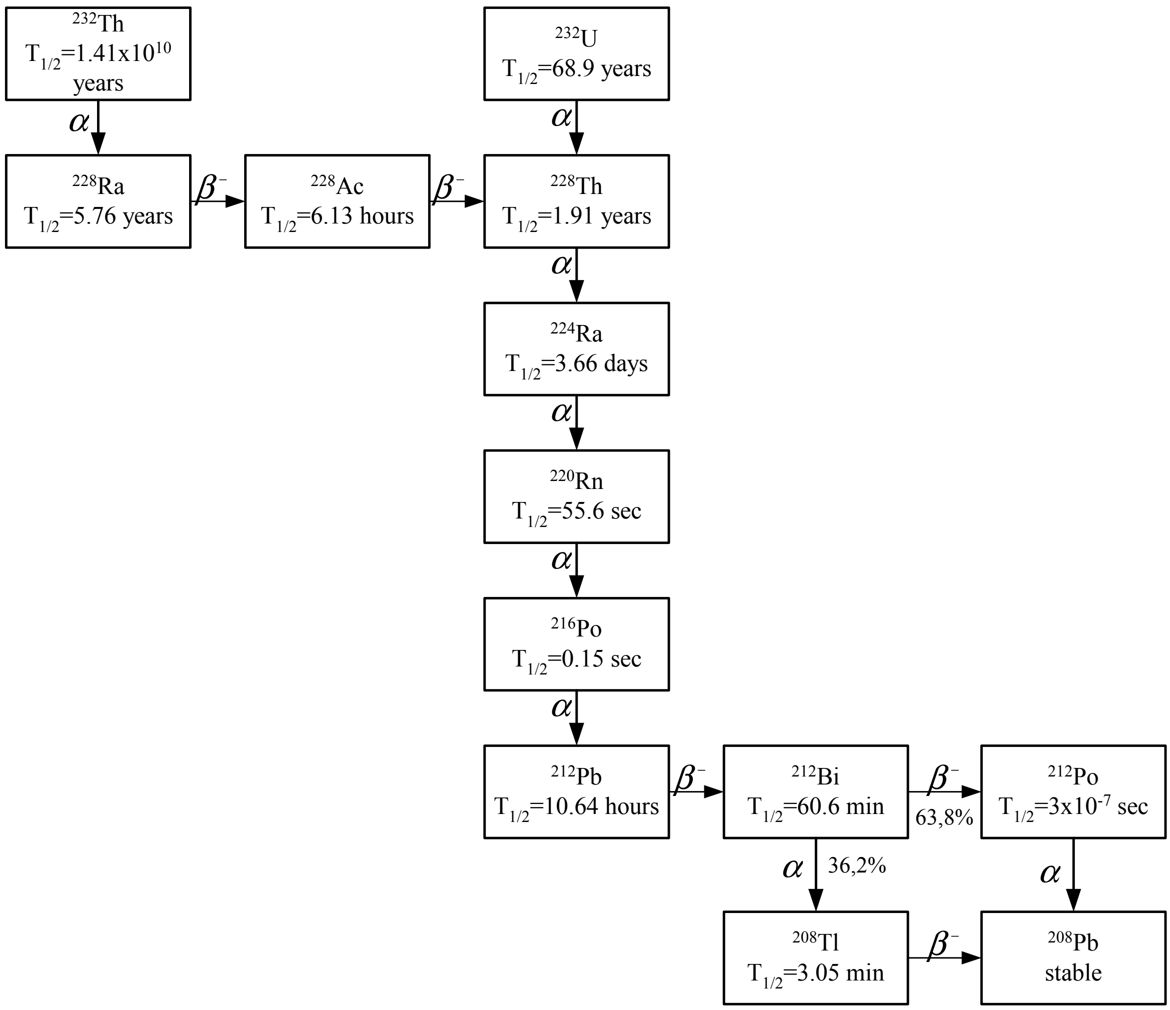


FIG. 1. Thorium radioactive chain.

It is important to note the inherent radiation barrier will be in action for full uranium mass because it is very difficult (if possible at all) to separate the generated ^{232}U from main uranium mass (even isotope separation technologies are applied). At the same time, separation of plutonium from uranium requires application of sophisticated chemical technology because, otherwise, α-activity of ^{232}U traces can cause neutron generation in (α, n)-reactions of light impurities. Even extraction of ^{228}Th (decay product of ^{232}U, see Fig. 1) from fuel can weaken only the proliferation protection barrier for a relatively short time period.

As ^{231}Pa has a large neutron capture cross-section in thermal range, this isotope can be considered, in principle, as a burnable absorber that allows to prolong time of fuel irradiation in the reactor core [5].

So, introduction of protactinium into nuclear fuel will provide the proliferation self-protection of fresh MOX-fuel in the closed nuclear fuel cycle (NFC). The formed protective profile of fuel in NFC may be considered as a complete one because it envelops both fresh and spent MOX-fuel.

3. NUMERICAL RESULTS FOR MOX-FUEL PROLIFERATION PROTECTION PROFILE

It is analyzed the closed LWR fuel cycle which is based on MOX-fuel with weapon-grade plutonium. LWR is operated at constant power level (mean specific heat generation - 110 kW/l). Value of fuel burn-up per one fuel campaign - 4% HM. Time of fuel management outside of LWR (outer fuel cycle) is adopted to be 5 years. This time period includes spent fuel cooling, spent fuel reprocessing and fresh fuel re-fabrication. It is supposed that chemical reprocessing of spent fuel provides complete extraction of fission products (FP) while ^{228}Th amongst other actinides remains in fuel for the next irradiation cycle.

It is adopted that achievement of the proliferation self-protection criterion requires that the inherent radiation barrier of fresh MOX-fuel would be at level of the Spent Fuel Standard with the same (5 years) cooling time. There are studied the effects of small protactinium additions to the loaded fuel on variation of MOX-fuel protection profile within one management cycle of the closed NFC. It was determined the value of protactinium content in MOX-fuel that satisfies the requirement presented above.

Radiation parameters of the fuel irradiated in LWR are calculated with application of the computer code package SCALE [6]. RED value at 30 cm-distance from MOX-FA is considered as a main characteristics of the inherent radiation barrier.

Variation of RED from MOX-FA with increasing the number of irradiation cycles is demonstrated in Table I (^{231}Pa content in fresh fuel is kept constant at level 0.5% HM). It can be seen that, cycle by cycle, the inherent radiation barrier of fresh MOX-fuel gradually increases. Three-fold recycle brings the proliferation self-protection of fresh fuel nearer to level of the Spent Fuel Standard from LWR with 5-year cooling time and exceeds it significantly at the equilibrium state. Note that transition into the equilibrium (for ^{232}U and ^{228}Th) regime can be carried out already at the first irradiation cycle by increasing initial ^{231}Pa content from 0.5% to 1.36% HM.

If we follow from the requirement that lethal dose is received within the minute range, then equilibrium ^{232}U content in fuel must be kept at a level of 0.3÷0.5% HM. In Table II, for a time moment t_{cool} = 5 years (this time corresponds to loading of recycled fuel into the reactor core), there are presented the evaluated equilibrium content of the isotopes which define a level of the MOX-fuel inherent radiation barrier under the adopted conditions of its irradiation and recycle: ^{231}Pa is a fertile isotope for generation of γ-radiation sources while ^{232}U and ^{228}Th are consecutive long-term and short-term, respectively, γ-radiation sources in fuel.

Combination of large neutron capture cross-section of ^{231}Pa and long half-life of ^{232}U as compared with time of fuel management cycle leads to rather high equilibrium ^{232}U content in MOX-fuel that satisfies the requirement of the lethal dose receiving within the minute range.

The proliferation protection of MOX-FA is conveniently to express in terms of time till the lethal dose receiving (T_{LD}) under unauthorized actions. Variations of the proliferation protection in the T_{LD} terms before and after irradiation in the reactor core for the established regime are demonstrated in Fig. 2. It can be seen that, after FP extraction, the proliferation protection of MOX-FA reduces twice remaining at level of three-minute T_{LD}.

R – radiochemical reprocessing of spent fuel; F – fuel fabrication.

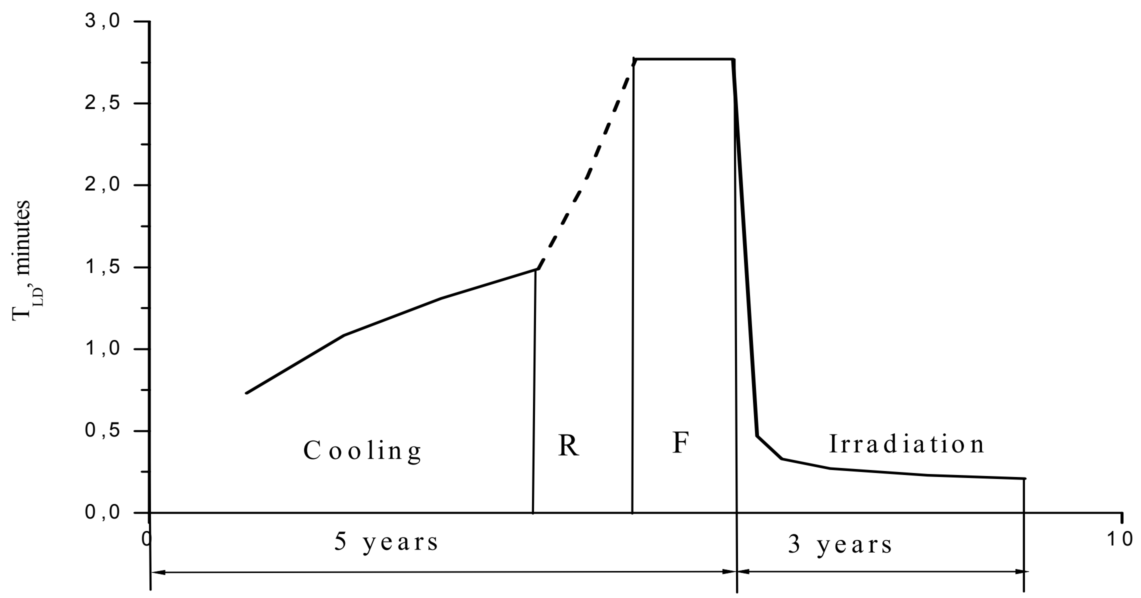


FIG. 2. Time till the lethal dose receiving (T_{LD}) at 30 cm-distance from MOX-FA (per one FA) at different NFC stages. T_{LD} at irradiation stage is evaluated for t_{cool} = 3 months.

Table I. Variation of the protection barrier per campaigns of recycled MOX-fuel (fresh fuel contains 0.5% ^{231}Pa at the beginning of each irradiation cycle)

	Cycle 1, after cooling t_{cool}=5 years	Cycle 2, after cooling t_{cool}=5 years	Cycle 3, after cooling t_{cool}=5 years	Equilibrium content	SFS t_{cool}=5 years
RED (rem/h r = 30 cm)	3240	5310	6580	9730	7570

Table II. Content of ^{231}Pa, ^{232}U, ^{228}Th in MOX-fuel under its multiple recycling in LWR (t_{cool} = 5 years)

	^{231}Pa	^{232}U	^{228}Th
Equilibrium content (% HM)	0.31	0.39	$1.07\ 10^{-2}$

Time dependence of content for the isotopes which define the inherent radiation barrier of MOX-fuel in the equilibrium cycle is presented in Fig. 3. For irradiation time, ^{231}Pa content in fuel reduces to 62% of initial value being constant further till its feeding which, thus, is equal to 38% of initial value for the equilibrium LWR operation regime. On the contrary, ^{232}U content for the equilibrium regime is nearly constant except of small increase of its concentration during irradiation in the reactor core and following small decrease due to natural decay during the cooling time.

In contrast to ^{232}U (that is generated under neutron irradiation of ^{231}Pa-containing fuel), ^{228}Th is a product of uranium α-decay with substantially shorter half-life ($T_{1/2}$=1.91 years). So, time dependence of ^{228}Th concentration in the fuel management cycle undergoes the marked changes. Presuming constant ^{232}U concentration (ρ_{U2}), dependence of ^{228}Th concentration (ρ_{28}) on irradiation time can be presented in the following analytical form:

$$\rho_{28}(t_{ir}) = \rho_{28}(0)\cdot e^{-\lambda'_{28}t_{ir}} + \frac{\lambda_{U2}\rho_{U2}}{\lambda'_{28}}\cdot\left(1-e^{-\lambda'_{28}t_{ir}}\right); \qquad (1)$$

where $\lambda'_{28} = \lambda_{28} + \langle\sigma_c\phi\rangle$, λ_{U2} , λ_{28} – decay constants of ^{232}U and ^{228}Th, respectively; $\langle\sigma_c\phi\rangle$ - neutron capture rate by ^{228}Th. Dependence of ^{228}Th concentration (ρ_{28}) on the cooling time can be presented in similar form (with neutron flux ϕ = 0). The formula (1) comprises of two terms, and relationship between them determines minimal ^{228}Th content that can be seen in Fig. 3.

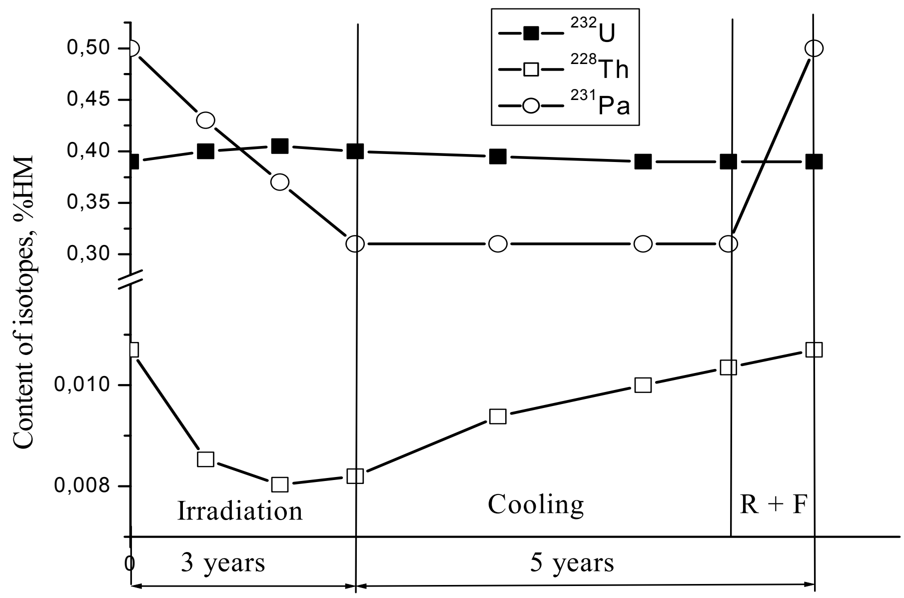


FIG. 3. Contents of ^{231}Pa, ^{232}U, ^{228}Th in a cycle of the equilibrium LWR operation regime.

It should be noted that the proliferation protection of fresh MOX-fuel is related with its γ-activity. So, fresh MOX-fuel management requires application of the promising fuel fabrication technologies (for example, technology of vibropacked fuel fabrication [7]) which, being essentially remote ones, permit the considered levels of fuel radioactivity.

4. APPLICATION OF CESIUM FOR COUNTERACTION AGAINST UNAUTHORIZED USE OF MOX-FUEL

Amongst γ-active radionuclides, there are considered mixtures of those nuclides which are fission products or products of their transformations under neutron irradiation. Isotope ^{137}Cs can be regarded, in the first turn, as the mostly long-lived γ-active FP. Let's analyse a possibility to arrange the proliferation protection of nuclear fuel using ^{137}Cs, and compare the proliferation protection with that obtained by addition of ^{231}Pa.

In contrast to protactinium, sufficiently large amounts of cesium are available in spent nuclear fuel (SNF) where ^{137}Cs is in a mixture with another cesium isotopes (^{133}Cs, ^{134}Cs, ^{135}Cs), typical for LWR-SNF.

Isotope ^{137}Cs is characterized by the following specific features: relatively long half-life ($T_{1/2}$=30 years) and, at the same time, low neutron capture cross-section (σ_c ~ 0.02 barn in LWR spectrum). So, theoretical value of equilibrium ^{137}Cs content is more than 10 times higher than that in typical LWR-SNF (0.23% HM) with fuel burn-up of 4% HM. The proliferation protection of ^{137}Cs-containing FA with ^{137}Cs contents close to equilibrium value exceeds significantly the proliferation protection criterion of the lethal dose receiving for a minute. Such a feature leads to the fact that, under multiple fuel recycle, only partial extraction of ^{137}Cs from SNF would be sufficient.

However, it should be noted that cesium oxide is less stable material than uranium and plutonium dioxides. In area of columnar grains, cesium can exist in form of separate atoms. Boiling temperature of cesium (678^0 C) is lower significantly than the working temperatures in power LWR. So, cesium can exist in a vapor phase and migrate in oxide fuel, thus causing the enhanced fuel-cladding interactions [8]. Therefore, in the present paper, cesium content twice as much as that in traditional LWR-SNF is adopted as an upper limit. Let's assess, under such a limitation, the proliferation protection of the reprocessed fuel at the mostly vulnerable NFC stages (fabrication and transportation to NPP) for MOX-fuel management regime described above (Section 3).

Variations in isotopic composition of cesium fraction in the fuel loaded into the reactor core are shown in Table III for the first two campaigns. Cesium content in fuel after the second campaign corresponds to the limiting value. So, for the next irradiation cycle, excess cesium quantity (≈36%) must be removed.

Multiple repetition of irradiation cycles with removal of excess cesium leads to equilibrium isotopic composition of cesium fraction. In contrast to another radioactive FP (Ce, Ru, etc.), share of ^{137}Cs in isotopic composition is kept at significant level (~ 30%) under multiple recycling.

The proliferation protection of fresh FA containing fuel for the second campaign corresponds to RED = 8650 rem/h at 30-cm distance from FA, i.e. time till the lethal dose receiving is about three minutes in vicinity of FA.

Table III. Variation of cesium isotopic composition for two consecutive fuel campaigns in LWR

	^{133}Cs	^{134}Cs	^{135}Cs	^{137}Cs
Initial loading, % HM	0.128	0	0.123	0.119
Composition after the first campaign (t_{cool}=5 years), % HM	0.360	0.006	0.358	0.330
Composition after the second campaign (t_{cool}=5 years), % HM	0.560	0.0105	0.580	0.500

Despite low share of ^{134}Cs in cesium isotopic composition (0.57%) and its short half-life ($T_{1/2}$ = 2.06 years), ^{134}Cs gives a substantial contribution to the proliferation protection level (about 40%). In contrast to ^{137}Cs and ^{134}Cs, isotopes ^{133}Cs and ^{135}Cs give no practical contributions into RED value because ^{133}Cs is a stable isotope while ^{135}Cs has a very long half-life ($T_{1/2}$ = 2.3 x 10^6 years).

It should be noted that keeping the proliferation protection of reprocessed fuel at level of 3-minute time till the lethal dose receiving is achieved for cesium content twice as much than that for traditional cesium content in SNF. The latter circumstance can be related with the difficulties following from low boiling temperature and corrosion activity of cesium.

5. CONCLUSIONS

MOX-fuel is one of the mostly promising fuel types for future nuclear power. So, MOX-FA, appearing at the international fuel market, must possess the proliferation self-protection property that makes it inaccessible for any unauthorized actions.

Analysis of the ways to the proliferation protection of MOX-fuel by small ^{231}Pa addition and comparison of this way with another options for giving MOX-fuel the proliferation self-protection property enable us to make the following conclusions:

(i) Unique nature of protactinium as a small addition to MOX-fuel is determined by the following properties:
  - Protactinium is available in the nature (uranium ore) as a long-lived mono-isotope ^{231}Pa.
  - Under neutron irradiation, ^{231}Pa is converted into ^{232}U, which is a long-term source of high-energy γ-radiation and practically non-separable from main fuel mass.
  - Essentially, ^{231}Pa is a high-quality burnable neutron absorber.

(ii) From the proliferation self-protection point of view, NFC closure with fuel recycle is a preferable option because it allows to arrange circulation of the inherent protected fuel.

(iii) Introduction of protactinium into MOX-fuel creates the inherent radiation barrier which is in action during full cycle of fuel management at the level corresponding to the accepted today criterion of the Spent Fuel Standard.

(iv) The analysed option of multiple MOX-fuel recycle with small addition of ^{231}Pa (0.2% HM) at each cycle demonstrates a possibility to reach the proliferation protection level of fresh MOX-fuel corresponding to once irradiated fuel with the same cooling time.

(v) Application of cesium from fission products (mainly, thanks to availability of isotope ^{137}Cs) can enable us to keep the inherent radiation barrier at the level of time till the lethal dose receiving within the minute range. However, high volatility and easy separation from fuel by calcination make cesium less attractive as a proliferation deterrent. The proliferation protection barrier based on ^{231}Pa addition into fuel is free of this disadvantage but requires a special technology for ^{231}Pa production. Apparently, combination of these two additions to MOX-fuel is able to enhance the inherent radiation barrier and to weaken shortcomings of both proliferation deterrents.

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STRATEGY OF NUCLEAR ENERGY DEVELOPMENT IN UKRAINE TILL 2030 AND FURTHER OUTLOOK: ROLE OF INNOVATIVE TECHNOLOGIES

L. LITVINSKY[a], O. PURTOV[a], A. ZENYUK[b]

[a] State Scientific - Engineering Center Of Monitoring Systems And Emergency Reacting (SSEC CSER)
Kiev, Ukraine

[b] Ministry of Fuel and Energy of Ukraine

Abstract. The basic purpose of Strategy is accepted effective and competitive operation of a nuclear power energy complex of Ukraine as a whole and its separate parts during all planning period. Strategy is elaborated by the Ministry of fuel and energy of Ukraine and National Atomic Generating Company "ENERGOATOM" with participation of leading scientific, design and engineering centers and institutes of Ukraine. For achievement of the Strategy purpose is necessary long-term mutual adjust planning of activity at all phases of a life cycle of nuclear objects and also activity on their support and maintenance. The role of innovative technologies for different directions of the activity is analyzed in the report.

1. INTRODUCTION

Strategy is elaborated by the Ministry of fuel and energy of Ukraine and National Atomic Generating Company "ENERGOATOM" with participation of leading scientific, design and engineering centers and institutes of Ukraine.

Today energy complex of Ukraine has total installed capacity 52,9 GW which includes 14 large thermal power plants (TPP) with installed capacity 36,4 GW (68,8%), 8 big hydro power plans (HPP) - 4,7 GW (8,9%) and 4 nuclear power plant (NPP) -11,8 GW (22,3%). Besides it there are autonomic power plants, not includes in electricity power grid of Ukraine with installed capacity 2,0 GW and block stations - 3,1 GW.

Ukraine borrows 8-th place in the world and 5-th place in Europe on acting nuclear units and their installed capacity. Dynamic of electricity production on NPP with comparison of total electricity production in Ukraine during 10 last years is presented on Figure 1. As you can see from the picture during long time nuclear energy complex (NEC) provides essential share of total electricity production in Ukraine (more 40%), that makes important condition of strategic planning of NEC development for stable economy development of whole country.

Results of following IAEA technical cooperation projects and other international projects were used for strategy development:

— IAEA TC project UKR/4/006 “Strategy of nuclear energy development”;
— IAEA TC project UKR/4/007 “Decommissioning NPP with WWER” etc.

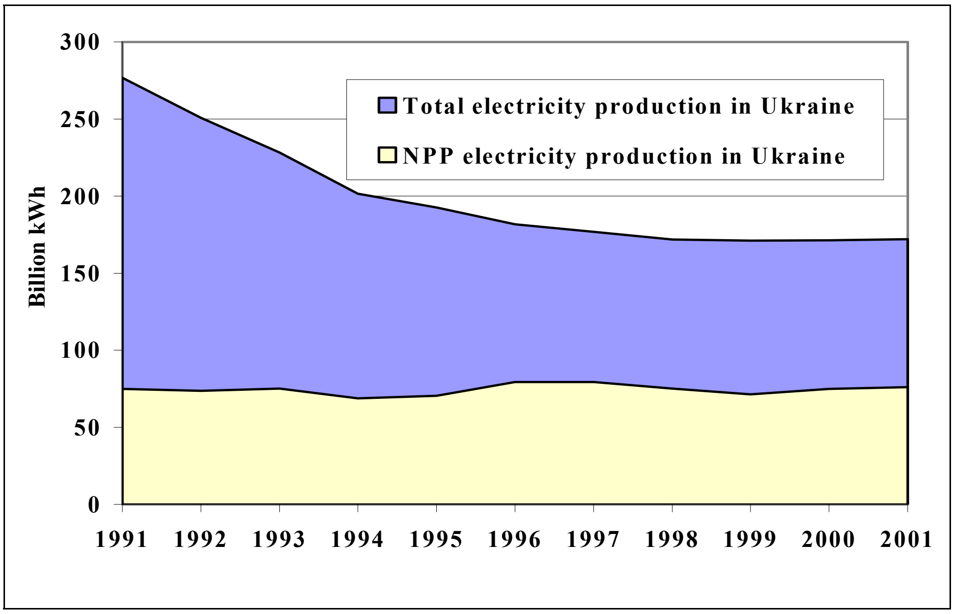

FIG. 1. Electricity generation in Ukraine during 1991 – 2001.

2. PLANNING OBJECTS

Main biggest objects of nuclear energy complex of Ukraine are:

— Zaporizhzhe NPP;
— Rivne NPP;
— Khmelnitsky NPP;
— Chornobyl NPP;
— South-Ukraine NPP;
— West mining-enrichment plant (WostGok, Zholtie Vody);
— Zirconium manufacture ventures.

Besides of enumerated objects NEC includes many manufactures, scientific, design-construction, and education plants so on. Summary information on NPP of Ukraine presented in Table I. Chornobyl NPP is decommissioning. Last of earlier exploited units with RBMK-1000 reactors was finally shut down on December 15, 2000.

On 4 operational NPP (Zaporizhzhe, Rivne, Khmelnitsky and South Ukrainian) in exploitation there are 13 units: 11 - with WWER-1000 reactor and 2 - with WWER-440. Operating organization all operational NPP of Ukraine is NAEK "ENERGOATOM".

The design lifetime of WWER reactor units is 30 years. By the foreseen initial projects the termination dates of exploitation of the operational NPP units of Ukraine are listed in Table I. Pursuant to the initial projects, since 2011 the measures on their decommissioning should start. At the same time world practice and the tentative estimations of a today's condition of the operational NPP units indicate a potential of extension of their exploitation for design lifetime.

Table I. Installed capacity units in NPP of Ukraine

NPP	Units	Reactor type	Installed capacity (MW)	Building start	Electricity production start	Design shut down time
Zaporizhzhe	1	WWER-1000/320	1000	04.1980	10.12.1984	10.12.2014
	2	WWER-1000/320	1000	04.1981	22.07.1985	22.07.2015
	3	WWER-1000/320	1000	04.1982	10.12.1986	10.12.2016
	4	WWER-1000/320	1000	01.1984	18.12.1987	18.12.2017
	5	WWER-1000/320	1000	07.1985	14.08.1989	14.08.2019
	6	WWER-1000/320	1000	06.1986	19.10.1995	19.10.2025
South-Ukrainian	1	WWER-1000/302	1000	03.1977	31.12.1982	31.12.2012
	2	WWER-1000/338	1000	10.1979	06.01.1985	06.01.2015
	3	WWER-1000/320	1000	02.1985	20.09.1989	20.09.2019
Rivne	1	WWER-440/213	402	08.1976	22.12.1980	22.12.2010
	2	WWER-440/213	416	10.1977	22.12.1981	22.12.2011
	3	WWER-1000/320	1000	02.1981	21.12.1986	21.12.2016
	4*	WWER-1000/320	1000	08.1986		
Chornobyl	1**	RBMK-1000	800	06.1972	26.09.1977	
	2**	RBMK-1000	1000	02.1973	21.12.1978	
	3***	RBMK-1000	1000	05.1977	03.12.1981	
Khmelnitzky	1	WWER-1000/320	1000	11.1981	22.12.1987	22.12.2017
	2*	WWER-1000/320	1000	02.1986		

* *construction finishing;* ** *decommissioning;* *** *shut down.*

As now and during all scheduled term till 2030 among NEC objects the installations at all stages of their lifetime cycle - from before design of surveys and designing (now - centralized storage of a spent nuclear fuel and radioactive waste, perspective nuclear units etc.) up to decommissioning (now - Chornobyl NPP, technology links and tailing pond of Pridneprovsky chemical plant etc.) will be submitted.

Thus, object of strategic planning is the operation of all nuclear power complex as a whole, including all conforming direction of activity:

— Control of the NEC subjects ;
— Safety regulation of NEC objects ;
— Managing at all stages of a life cycle of NEC objects ;
— Maintenance of NEC objects by consumed resources;
— Treatment with NEC technological waste.

3. PURPOSE, CONDITIONS AND DIRECTIONS OF STRATEGIC PLANNING

The main purpose of strategic planning is the maintenance economically of effective and competitive operation of a nuclear power complex of Ukraine as a whole and its separate parts during all scheduled term till 2030 and on a further outlook.

Determining conditions of strategic planning are:

— Unconditional observance of all standards and requirements on safety of objects of nuclear power and limitation of their influencing on the population and environment (priority of safety);
— Maintenance of national security of Ukraine on fuel-energy indexes (priority of national concerns);
— Maintenance of an effective utilization before the enclosed costs (priority of correctness of investments);
— Maintenance of continuity of operation of a nuclear power complex of Ukraine outside the scheduled term (priority of stability hereafter);
— Minimization outside the scheduled term negative economical, social, ecological etc. consequences from NEC operation during the scheduled term (priority of hardening of following generation).

For achievement of the main purpose it is required long-term conforming planning economically to effective activity at all stages of a life cycle of objects NEC, and also activity on their support and maintenance. Outgoing from this, reference directions of strategic planning are determined:

— Normative - legal maintenance of development NEC;
— Exploitation operational NPP;
— Lifetime extension of NPP;
— Development of domestic parts of a nuclear - fuel cycle;
— The treatment with a spent nuclear fuel;
— The treatment with a radioactive waste;
— Decommissioning NPP of Ukraine;
— Perspective construction in nuclear power engineering;
— Industrial and technological maintenance NEC;
— Scientific - engineering and design support NEC;
— Social and personnel problems of development NEC.

The degree of interaction interference of activity on listed directions is routine in a Fig. 2.

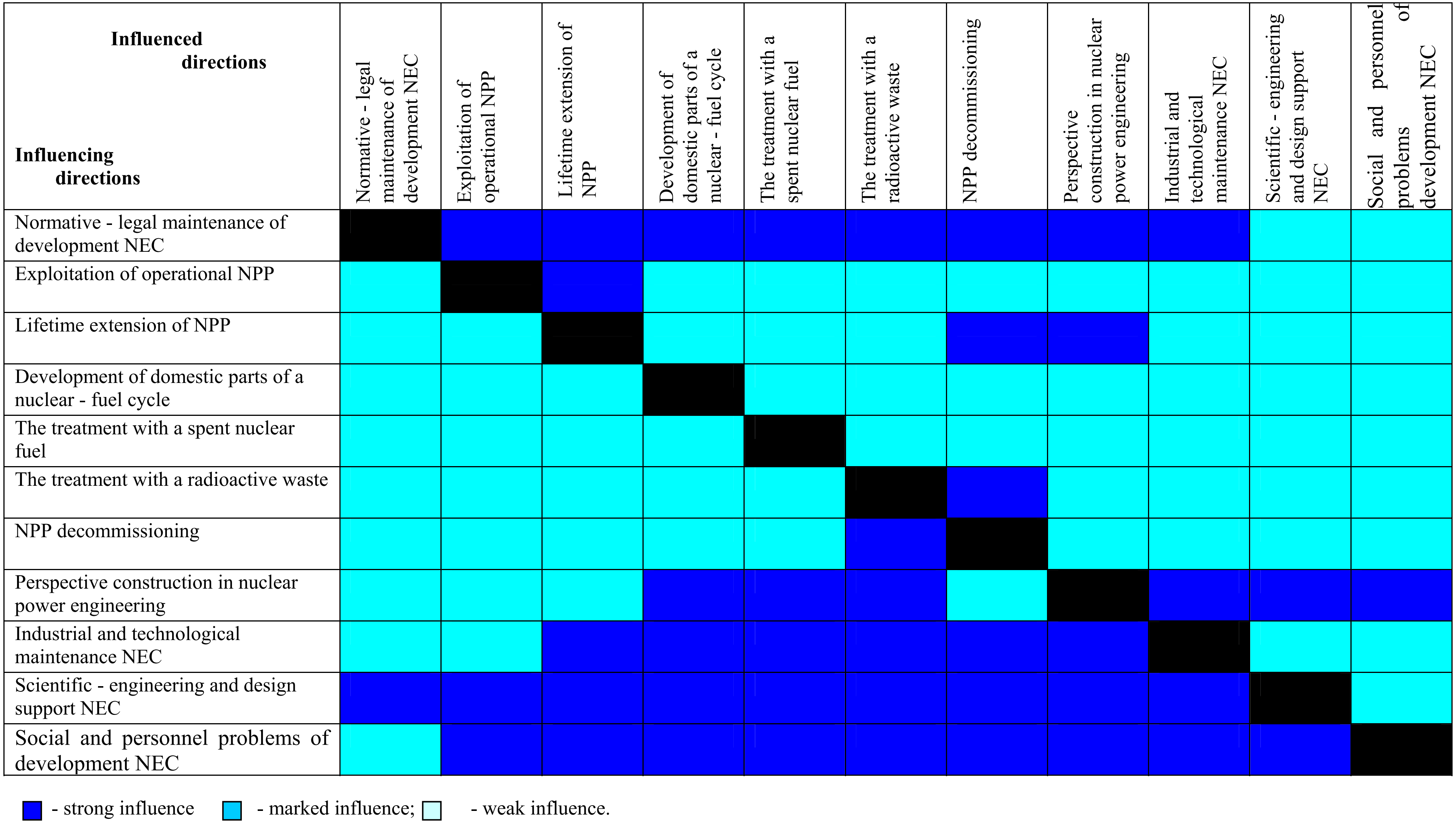

FIG. 2. Inter influence of main directions of NEC activity.

4. ROLE OF INNOVATIVE TECHNOLOGIES

Practically all aspects of activity on each of directions are connected with an implementation of innovative technologies. However feature of development NEC is the long-term exploitation (30 years and more) of already built objects. As on the moment of the start in exploitation any of object all requirements on safety are already ensured its modernization and renovation during design lifetime has restricted nature. Therefore role innovative technologies essentially is various for miscellaneous stages of a life cycle of NEC objects.

Except for design and construction stages of new object, where the contribution innovative technologies makes near 100 %, quantitatively to define this contribution at other stages of a life cycle of NEC objects it is not obviously possible. Qualitatively role innovative technologies at these stages is arranged following by a mode (in decreasing order):

— Designing and construction;
— Lifetime extension for terms foreseen initial project;
— Exploitation beginning;
— Decommissioning;
— Exploitation.

The implementation of innovative technologies for miscellaneous directions of strategic planning directly is connected to scientific - engineering and designs support. As it is visible from a Figure 2, except for social and personnel problems of NEC development, all other directions are subject to strong influencing of this direction.

5. SCENARIOS OF DEVELOPMENT OF A NUCLEAR ENERGY COMPLEX OF UKRAINE AND CONTRIBUTION OF INNOVATIVE TECHNOLOGIES IN ELECTRICITY PRODUCTION

Key directions determining the basic difference of the possible scenarios of NEC development in Ukraine till 2030 and a further outlook, are those directions, which one are essentially determined by an implementation of innovative technologies:

— Perspective construction in nuclear power engineering;
— Lifetime extension of exploitation of operational NPP units.

Now for NPP in Ukraine there is a depleting input information indispensable for the unequivocal forecast of terms, the exploitation of the operational nuclear units can be prolonged to which one. Outgoing from tentative estimations and world in-service experience of a water-water type reactor, three optional versions of extension - on 10-15 years, for 10 years and on 5-10 of years outside 30-year's term, foreseen initial projects are determined.

Terms of the introducing in exploitation of new nuclear units are determined by the following requirements:

— to cover of a deficit of base capacities at decommissioning of the operational units;
— by absence of dead excess of installed capacities;
— by reasonable limitation of investments and optimum schedule of their expences.

Outgoing from above-stated, three scenarios of NEC development till 2030 and further outlook - scenario - 1, scenario - 2 and scenario - 3 are determined. The scenario - 1 conforming to maximum extension of exploitation operational АЭС, guesses:

— the effective lifetime of nuclear units will be prolonged on 10-15 of years;
— a total number of new units by conditional power 1000 MW, which one will be put into operation till 2030 are 7 units (including RNPP-4 and KhNPP-2);
— in 2030 in a construction condition there will be 4 units.

The scenario - 2 guesses:

— the effective lifetime of all nuclear units will be prolonged to 10 years;
— a total number of alternating units by conditional capacity 1000 MW, which one will be put into operation till 2030 are 9 units;
— in 2030 in a condition of construction there will be 3 units.

The scenario - 3 guesses:

— the effective lifetime of all nuclear units will be prolonged on 5-10 of years;
— a total number of alternating units by conditional capacity 1000 MW, which one will be put into operation in the term 2006-2030 are 11 units;
— in 2030 in a condition of construction there will be 3 units.

Dynamics of change of an installed capacity on NPP for 3 scenarios is routine in a Figure 3-5. Except for two nuclear units such as WWER-1000 (unit 2 Khmelnitskiy NPP and unit 4 Rivne NPP), the construction which one will be finished in nearest 2 years, the creation of remaining new units and other NEC objects is planned on the basis of future perspective advanced solutions. Therefore, as was marked above, the production of the electricity by new NPP capacity will be determined predominantly by implementation of innovative technologies.

Largely innovative technologies will be determined also and share of the electricity, producing by units exploited outside terms foreseen initial projects (Figure 3-5).

In summary authors express a profound gratitude to P.Trampus, M.Laraia for the considerable help at generalization of world experience of forward planning in nuclear power engineering.

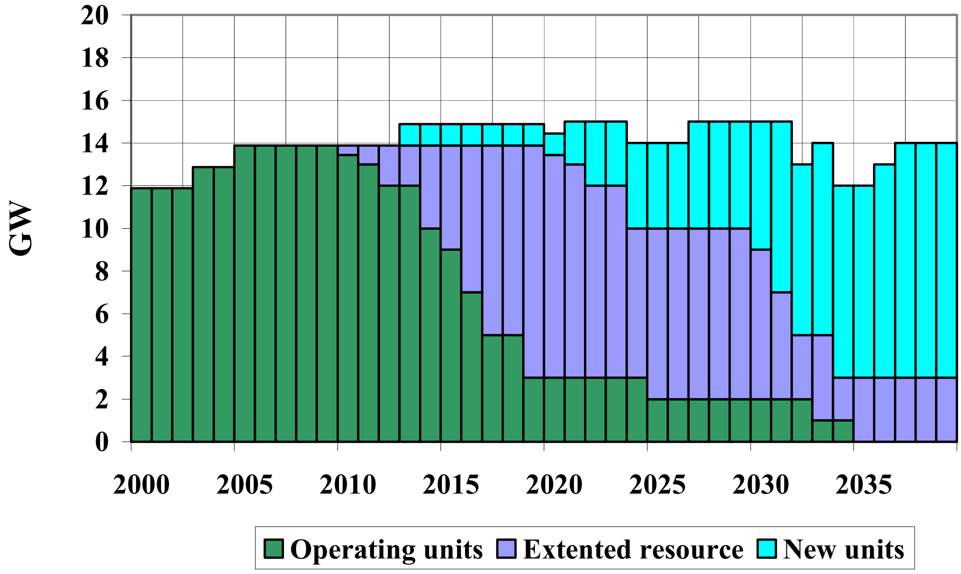

FIG. 3. Scenario-1. Dynamics of change of an installed capacity on NPP (GW).

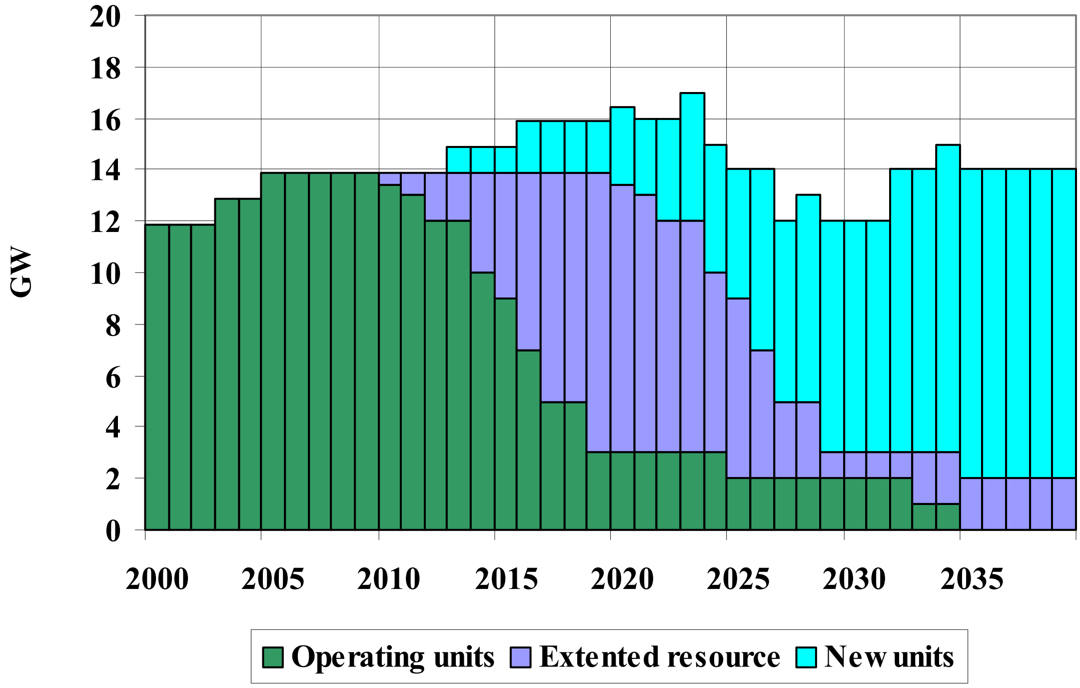

FIG. 4. Scenario-2. Dynamics of change of an installed capacity on NPP (GW).

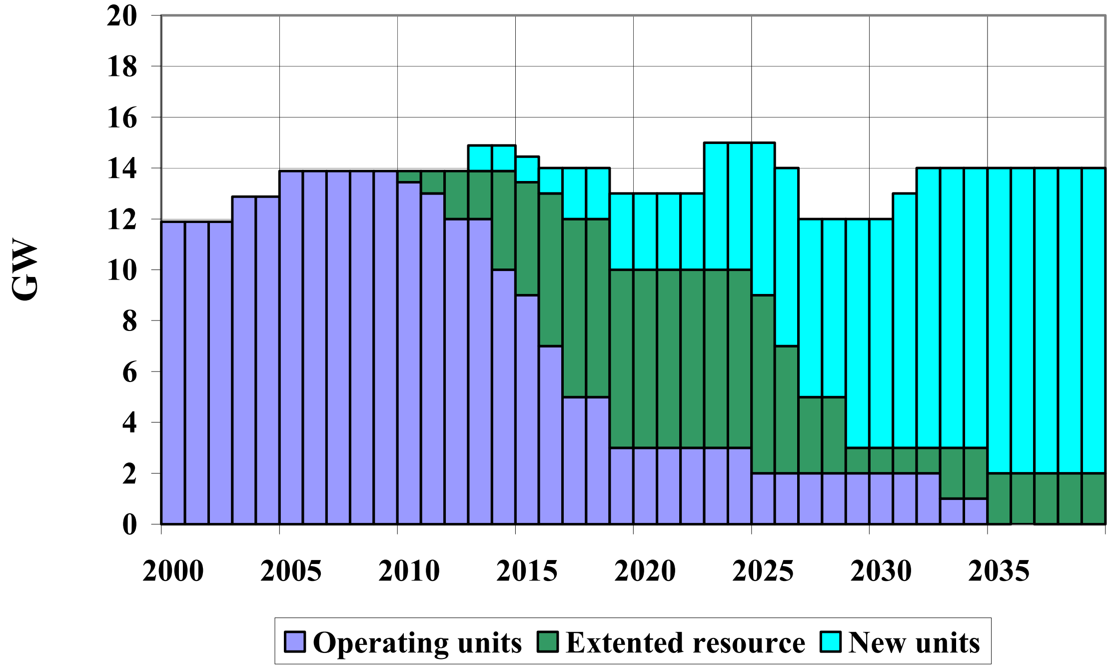

FIG. 5. Scenario-3. Dynamics of change of an installed capacity on NPP (GW).

VVER-440 AS THE FIRST-TIER REACTOR

P. DARILEK[a], V. NECAS[b], V. SEBIAN[b]

[a] Department of Nuclear Design and Fuel Management
Engineering, Design and Research Organization
Trnava, Slovakia

[b] Department of Nuclear Physics and Technology
Faculty of Electrical Engineering and Information Technology
Slovak University of Technology
Bratislava, Slovakia

Abstract. Exploitation of VVER-440 as the first-tier reactor at the multi-tier transmutation system is suggested. Influence of appropriate modification of fuel management on VVER-440 fuel cycle back-end is analysed numerically by the spectral code HELIOS. Theoretical model is based on preliminary evaluation of VVER-440 transmutation potential. Partially closed fuel cycles of the reactor VVER-440 are characterised. Two types of so-called "combined fuel assembly" with "transmutation pins" are described. Non-uranium fuel pins are used at the first type and inert matrix fuel is typical for the second one. Quantitative results of developed cycles are compared with "classical" open fuel cycle.

1. INTRODUCTION

"Ecological" nuclear energy production has good changes to cover growing electricity demands. Main environmental problem connected with nuclear energy exploitation is spent fuel management, discharged annually from some 400 commercial nuclear power plants now in operation.

The management of spent fuel should ensure that the biosphere is protected under economically acceptable conditions without entailing unfavourable short-term consequences and the public must be convinced of the effectiveness of the methods. Since the spent fuel contains very long-lived radio-nuclides some protection is required for at least 100 000 years. Two ways are possible:

— waiting for the natural decay of the radioactive elements isolated physically from the biosphere by installing successive barriers at a suitable depth in the ground. This strategy leads to deep geological disposal;
— exploitation of nuclear reactions that will transmute the very long-lived wastes into less radioactive or shorter-lived products.

PWR reactors are often taken into account for performance of spent fuel transmutation. More detailed evaluation of VVER-440 reactor transmutation potential [1] is given at following paragraphs.

2. MULTI-TIER TRANSMUTATION

Reactors PWR – the most frequent nuclear energy sources and other reactor types and concepts in exploitation or under development can serve as first “natural” candidates for performance of (at least partial) spent fuel transmutation. Such evolution of nuclear technology can profit from gathered exploitational and experimental experience. Accelerator Driven Systems (ADS), Molten Salt Reactors (MSR) and other revolutionary technologies on the other side can supply high enough neutron densities and can reach sufficient transmutation efficiency. But such new technology needs decades of expensive research and development. Multi-tier transmutation system [2] is natural connection of mentioned approaches. It performs combination of transmutation effects as follows:

- much of plutonium and perhaps other problematic materials can be consumed in power reactors
- (Tier 1 with thermal spectrum) and more actinides and fission products can be destroyed at accelerator – based or other transmutation systems (Tier 2 – fast spectrum).

Such complex multi-tier system can reach objectives as follows:

- to improve long-term public safety by reduction of spent fuel radio-toxicity and future inhabitants peak doses;
- to provide benefits to the repository program by reduction of heat producing material inventory and spent fuel mass and by minimisation of criticality risk;
- to reduce the proliferation risk from plutonium in commercial spent fuel;
- to improve prospects for nuclear power by waste management problem solution.

Partial first tier transmutation can be performed also at existing or new light water reactors (LWR′s) [3,4]. Pu, minor actinides and even some long–lived fission products (LLFP) can be burned there partially. This first tier concept can exhibit some advantages for second tier by minimisation of reactivity swing during second tier cycle by minimisation of Pu content in second tier fuel or by reduction of the amount of material sent to the second tier.

3. VVER-440 AT FIRST TIER

VVER reactor, that are in exploitation in Russia and some other countries in Europe (and Asia) can be considered as specific type of LWR reactors. Main difference in comparison with LWR is hexagonal shape of FA′s and triangular net for FA′s and pins.

Back-end solution depends also on economy power of country, solving the problem. At small nuclear economy with high share of nuclear electricity (Slovakia with 5 million people and more than 50 % electricity form six VVER-440 reactors is typical example) high-level waste consists of burned FA′s and some other materials from NPP operation (no problems with military abuse of Pu). Partitioning and transmutation is contemporary progressive back-end solution. But economical potential does not allow individual development of some revolutionary back-end technology as ADS. Selected method should be as simple as possible in order to facilitate its development and implementation. The only realistic way is to participate on international development effort and explore existing equipment and experience.

Reactor VVER-440 was taken into account as safe, reliable and long time exploited candidate for first tier reactor and very simple separation process was supposed at the first stage. First evaluation of transmutation fuel cycles of VVER-440 reactor with non-uranium (NU) or inert matrix (IM) fuel in so-called combined assemblies (CA's) is described in following paragraphs.

4. COMBINED FUEL ASSEMBLY

CA (Figure 1) is a model of special transmutation fuel assembly. Its purpose is to transmute the transuranic elements (TRU) (and fission products (FP)). Geometry of CA is the same as geometry of the original fuel assembly used regularly at the VVER-440 reactor (in the following text labeled as VVER FA).

The only difference between CA and VVER FA is material composition. There are 126 uranium fuel pins in VVER FA but 96 U-pins (or 120 U – pins) and 30 transmutation pins with inert matrix fuel (TP_{IM}) (or 6 transmutation pins with non-uranium fuel – TP_{NU}) in CA. The purpose of T-pins is transmutation of TRU (and FP) without breeding of another TRU. TP_{IM} consists of transuranic elements oxides ($TRUO_2$) and zirconia (ZrO_2). The latter component is used as an inert matrix material. TP_{NU} consists of $TRUO_2$ and fission products (FPO_2). Composition of U-pins in the CA is the same as in VVER FA with enrichment 4.0 w% (or 5.0 w%) of ^{235}U.

5. EQUILIBRIUM FUEL CYCLE OF CA

CA equilibrium fuel cycle (EFC) [5] preparation is shown on Figure 2. Fresh CA or fresh VVER FA (at first cycle) is burned-up to 40 MWd/kgHM in the core of VVER-440 and cooled for 5 years at the reactor pool. Partitioning and transmutation process integrated with fuel cycle can be divided into five stages as follows:

1. Burned U-pins and T-pins are separated (96 UP and 30 TP_{IM} or 120 UP and 6 TP_{NU} from CA, 126 UP from VVER FA).
2. Uranium is separated from burned U-pins and is manipulated as low level waste (LLW) or can be used as a material for fresh U-pins preparation.

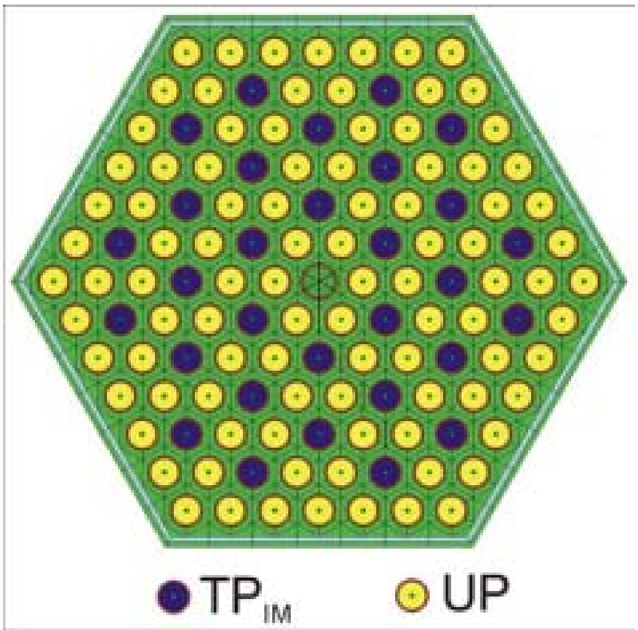

FIG. 1. The model of CA_{IM}.

3. TRU and FP - rest of burned U-pins after uranium separation – are concentrated to the T-pins and together with fresh U-pins are used for fresh CA's preparation.
4. T-pins are not reprocessed, they are regarded and manipulated as high level waste (HLW) and wait for second tier. Second-tier facility (ADS, MSR, fast breeder reactor – FBR, ...) should use fast spectrum of neutrons to reach the maximum transmutation effect.
5. Fresh CA is loaded into the VVER-440 core, burned-up to 40 MWd/kgHM, cooled for 5 years and cycle comes back to the stage 1.

VVER FA and CA burn-up was calculated by spectral code HELIOS [6]. Described fuel cycle was reprated 10-times and then mass concentrations of FP and TRU were analysed. The positive effect of this way of transmutation on assembly level is a general decrease of ^{239}Pu concentration at T-pins – see Figure 3. On the other hand, the negative effect is concentration increase of some TRU and FP, for example ^{246}Cm – see Figure 4. Quantitative effects on the core level are given in following paragraphs.

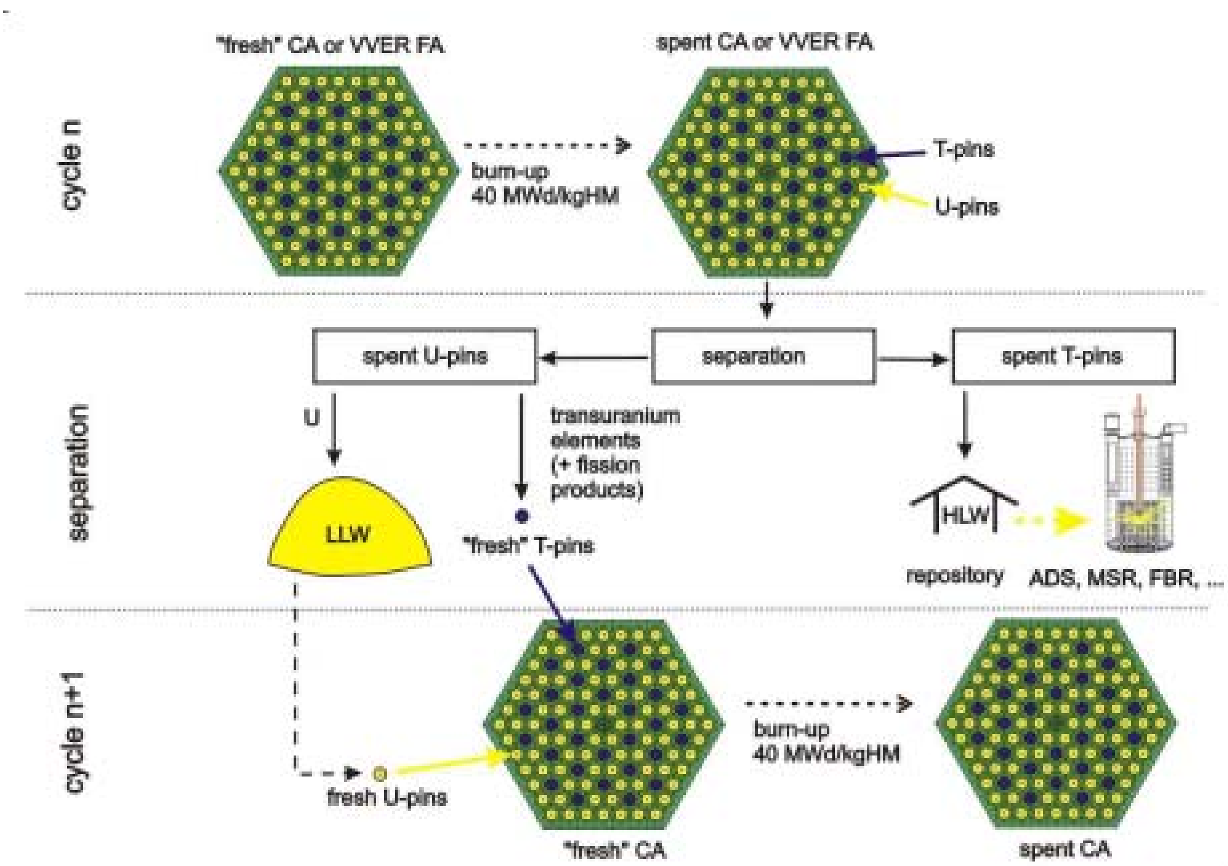

FIG. 2. Equilibrium fuel cycle with CA.

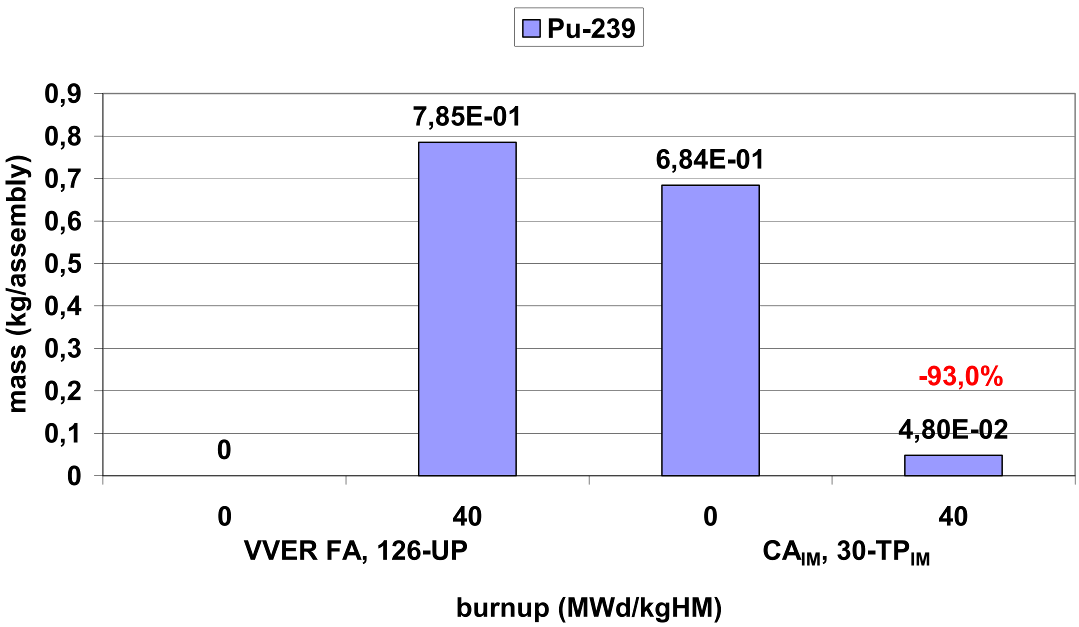

FIG. 3. Mass comparison of ^{239}Pu after burn-up.

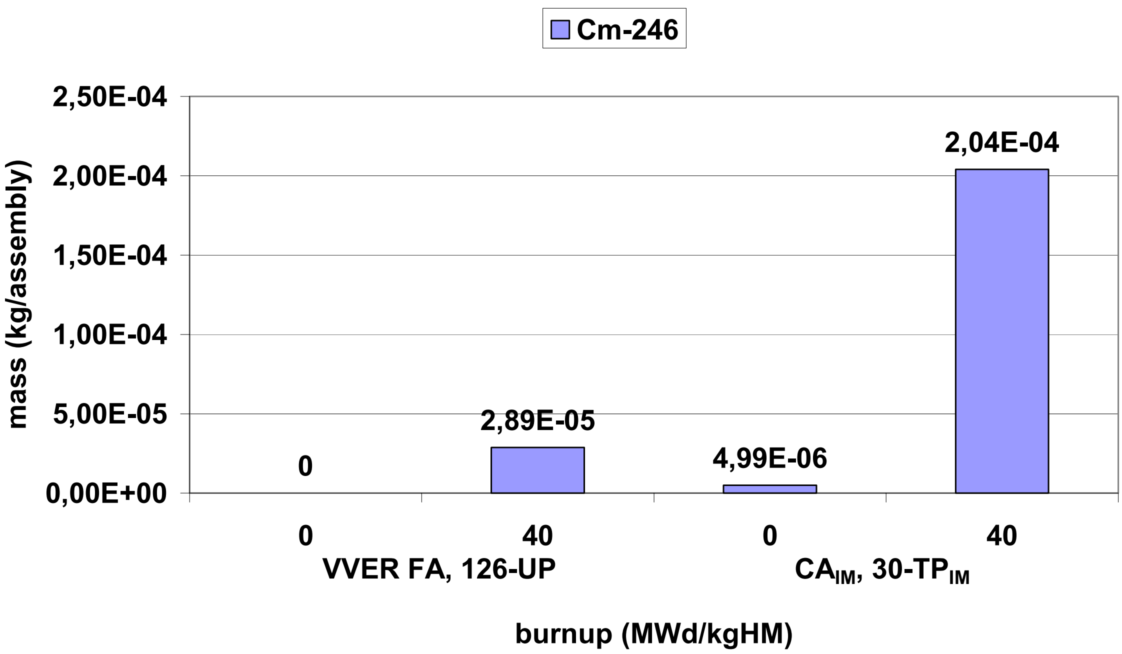

FIG. 4. Mass comparison of ^{246}Cm after burn-up.

6. PARTIALLY CLOSED FUEL CYCLE OF VVER-440

Partially closed equilibrium fuel cycle (PCFC) model of VVER-440 is shown on Figure 5. Material flow can be described as follows (values in brackets are valid for $PCFC_{NU}$):

87 fuel assemblies are discharged after 40 MWd/kgHM burn-up (13 VVER FA′s and 74 CA′s) from the VVER-440 core. U is separated from 8742 U-pins (or 10 518 U-pins). TRU (or TRU and FP) from the rest are concentrated and 2220 TP_{IM} with inert matrix (or 444 TP_{NU}) are produced. 7104 fresh U-pins (or 8880 fresh U-pins) are added to the fresh T-pins and 74 fresh CA′s are created. By addition of 13 fresh VVER FA′s fresh fuel batch is completed and loaded into the VVER-440 reactor core. Spent 2220 TP_{IM} (or 444 TP_{NU}) are moved to the storage to wait for second-tier of multi-tier transmutation system. Uranium separated from spent U-pins (~8015 kg or ~9571 kg) is manipulated as LLW or used for fresh U-pins preparation.

Large contrast of quantity of HLW released from current OFC of VVER-440 reactor and from the PCFC model can be seen in Figure 6. There is more than 10 t of HLW at OFC but less than 1.4 t (or 0.5 t) of HLW and **no uranium** at PCFC's. Uranium is considered as LLW there. Other advantage of PCFC' is saving of about 3.5 (or more than 17) VVER FA′s per fuel cycle as a consequence of reusing of TRU that were bred in U-pins at previous cycle (at 444 or 2220 T-pins - see Fig.5).

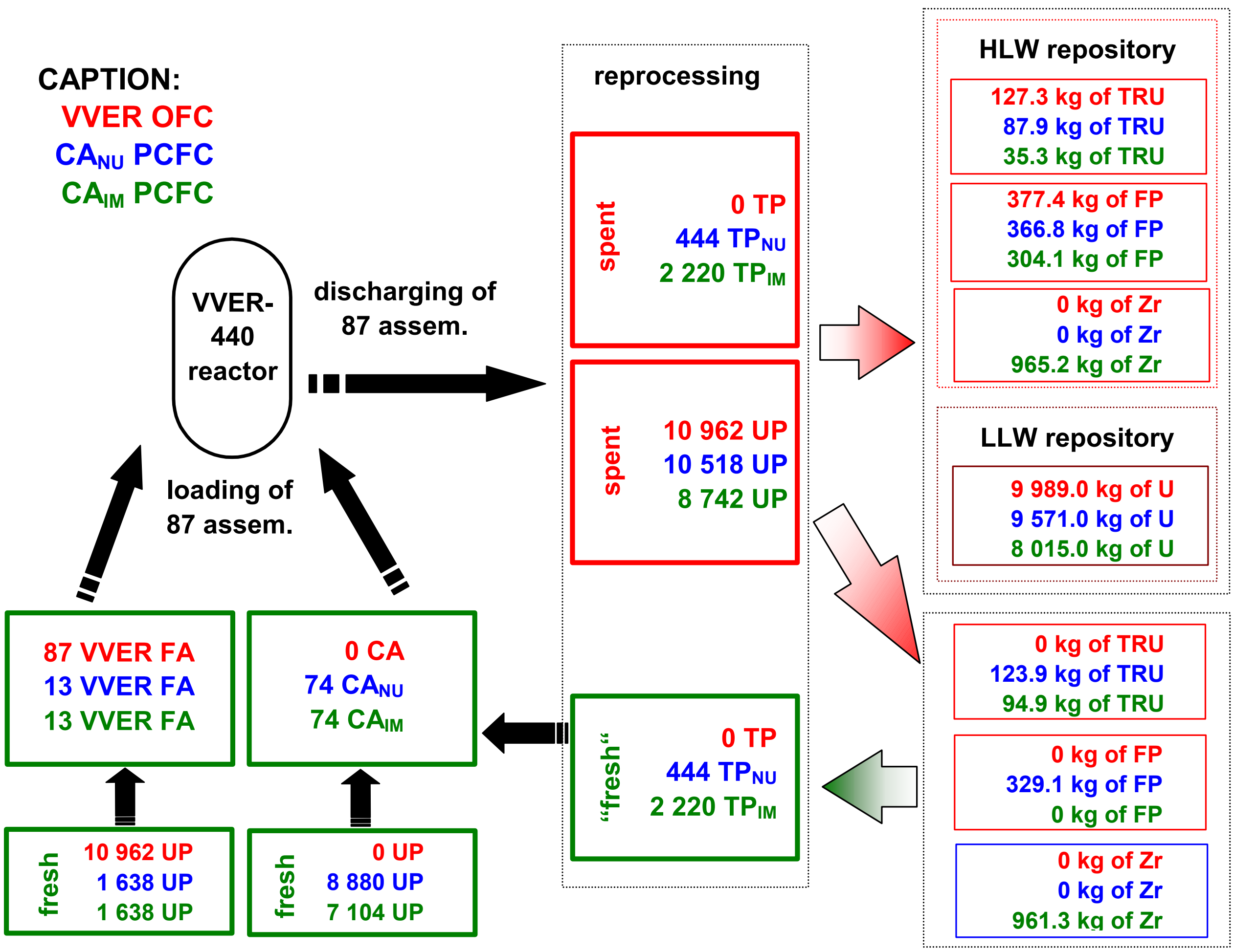

FIG. 5. Open and partially closed fuel cycles.

mass (kg)	OFC		$PCFC_{HU}$		$PCFC_{IM}$	
	HLW repository	LLW repository	HLW repository	LLW repository	HLW repository	LLW repository
FP	377.36	0	366.83	0	304.08	0
TRU	127.25	0	87.86	0	35.31	0
U	9989.02	0	0	9578.14	0	8014.97
Zr	0	0	0	0	965.17	0
SUM	10492.29	0	454.69	9578.14	1304.56	8014.97

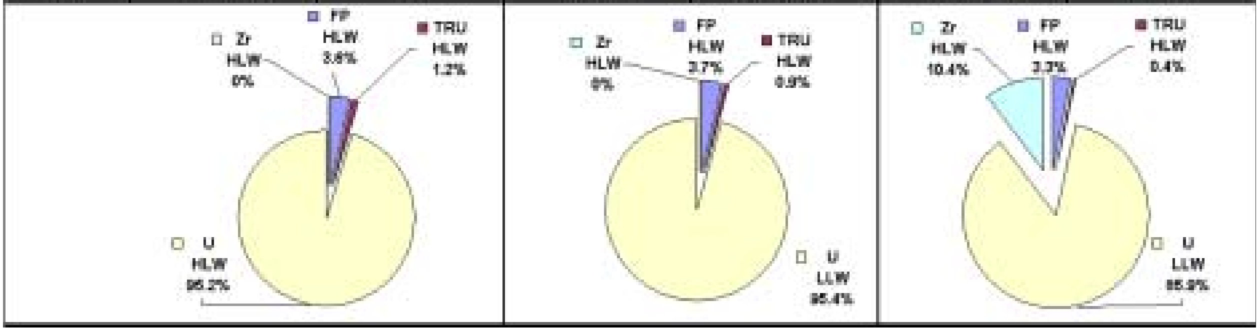

FIG. 6. Comparison of material flows at OFC and PCFC's.

Fig. 7 and 8 show a comparison of selected radionuclides flow at the current OFC and the suggested $PCFC_{IM}$. For example, mass of ^{239}Pu is lower by 94.8% in comparison with OFC. However, not all transuranic elements are well transmuted in the thermal spectrum of neutrons, for example mass of ^{246}Cm is higher by 5× at $PCFC_{IM}$.

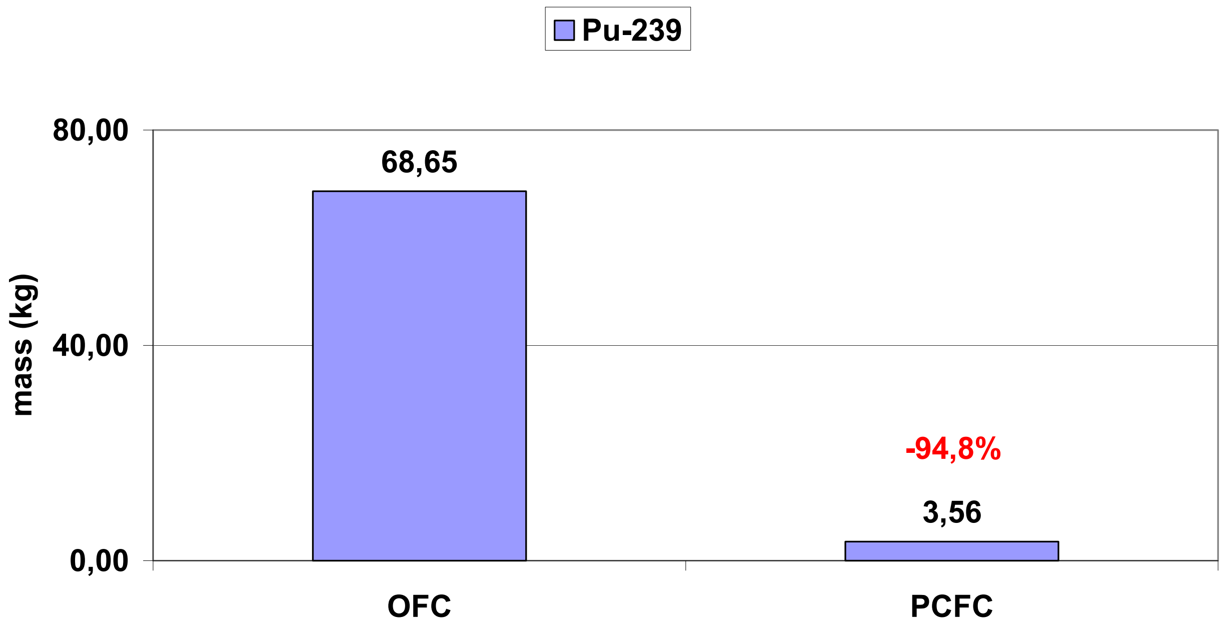

FIG. 7. The mass comparison of ^{239}Pu in cycles.

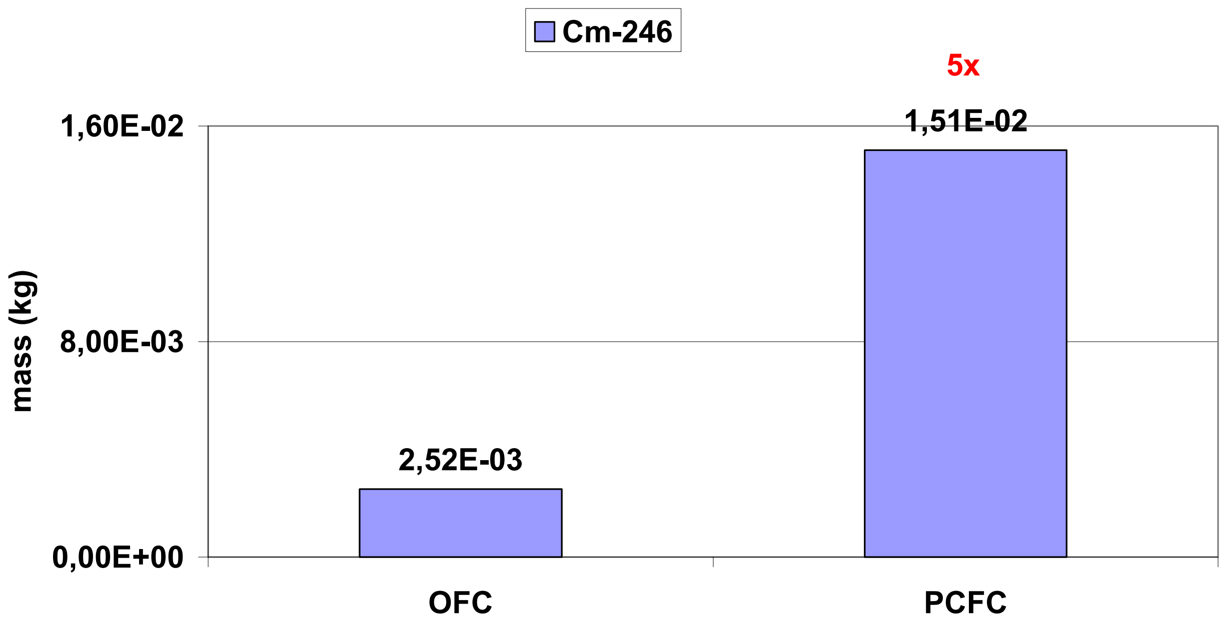

FIG. 8. The mass comparison of ^{246}Cm in cycles.

7. CONCLUSION

Effective solution of fuel cycle back-end as a condition for nuclear energy production development was pointed out. Multi-tier transmutation system was outlined out as a progressive combination of revolutionary concepts (ADS, MSR, FBR, ...) with evolution of existing technology - namely LWR′s. Reactor VVER-440 was suggested as first-tier reactor at small nuclear economies based on VVER type reactors.

Introduction of transmutation process into the VVER-440 fuel cycle based on combined fuel assembly with transmutation pins was described. Resulting model of partially closed VVER-440 fuel cycle analysed by spectral code HELIOS exhibited significant advantages in comparison with common open fuel cycle. Minimal positive effects are as follows:

— high-level waste flow to the deep repository was reduced 8-times
— amount of resulting TRU was of 30 % smaller
— fresh fuel economy was reached by replacing of 444 fresh U-pins (equivalent of more than 3 fuel assemblies) by “fresh” T-pins.

Analytical results show, that VVER-440 is reasonable candidate on first-tier reactor at multi-tier transmutation system.

This project has been the part of SR R&DP 1800 and supported partially by the Slovak Grant Agency for Science through the grant No. 1/0285/03.

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REQUIREMENTS AND EXPECTATIONS FROM INNOVATIVE NUCLEAR REACTORS: TURKEY'S PERSPECTIVE

A.B. BOLME, A. TANRIKUT, O. GUNDUZ, S. KOSE

Turkish Atomic Energy Authority
Technology Department
Ankara, Turkey

Abstract. After the postponement of Akkuyu NPP project in 2000 due to economical reasons, Turkish Atomic Energy Authority (TAEK) commenced a review of national nuclear policy of the country. In the announcement of the postponement Government stated that Turkey's interest to nuclear reactors would continue and Turkey might utilize new generation nuclear reactors in the future. It was also stated that Turkey is willing to participate and contribute to the development of new reactors. In view of these statements and recent developments in the energy sector, TAEK outlined the requirements and expectations regarding new nuclear reactors and decided to participate in some ongoing international studies for the development of innovative reactors. Some of the requirements determined by TAEK are; a) low capital and low electricity generation costs; b) short construction period; c) short licensing period; d) enhanced safety; e) utilization of proven technology; f) environmentally friendly design; g) suitability for public acceptance; h) utilization of indigenous resources; i) and suitability for hydrogen production, desalination and process heat. Studies for the determination of new nuclear policy are continuing.

1. BACKGROUND

1.1. Early studies

Studies to build a nuclear power plant in Turkey were started in 1965. Between 1967-1970 A feasibility study was made by a foreign consultant company to build a 300-400 MWe NPP. The NPP would have been in operation in 1977. Because of the problems mainly relating the site selection the project could not come into life.

In 1973, Turkish Electricity Authority (TEK) decided to build an 80 MWe prototype plant. This project was cancelled in 1974 due to reason that the project might delay the construction of a greater capacity nuclear power plant. Instead of this prototype plant, TEK has decided to build a 600 MWe NPP in southern Turkey.

1.2. First bid

Site selection studies have been made in 1974-1975 and Gülnar-Akkuyu location in southern coast of Turkey was found suitable for the construction of the first NPP. In 1976, Site license for Akkuyu was granted by Atomic Energy Commission. The bid was prepared in 1977 and ASEA-ATOM and STAL-LAVAL companies awarded as the best bidders. Contract negotiations continued until 1980. However in 1980 due to Swedish Government's decision to withdraw the loan guarantee project has been cancelled.

1.3. Second bid

In 1980, three companies have been awarded to build four nuclear power plants (1 unit CANDU (AECL) and 1 unit PWR (KWU) in Akkuyu and 2 units BWR (GE) in Sinop). Due to Turkey's suggestion of "Build-Operate-Transfer (BOT) model", KWU resigned from the bid. Although AECL accepted the BOT model, insisted upon the governmental guarantee on the BOT credit. Turkish government refused to give the guarantee, thus the project was cancelled.

1.4. Third bid

In 1992, Ministry of Energy and Natural Resources stated in a report submitted to the Government that without the installation of new energy resources before 2010, the country would face an energy crisis, suggesting that nuclear energy generation should be considered as an option. in 1993, The High Council of Science and Technology established the nuclear electricity generation as the 3rd highest priority project of the country. In view of this decision, Turkish Electricity Generation and Transmission Company (TEAŞ) included the NPP project in its 1993 investment program. In 1995, TEAŞ selected the Korean KAERI as the consultant for the preparation of the bid specifications. The Third Bid was started in 1996. Three companies made proposals: AECL, NPI and Westinghouse. After a series of delays the Government decided to postpone the project in 2000.

1.5. Reasons of postponement

In spite of the fact that nuclear energy contribution was planned to be 9,000 MW_e by the year 2020 (9% of the expected total generation) and there was a strong intension of the Government to install the fist NPP in Akkuyu, the Government has decided to postpone the Akkuyu NPP project, following the meeting of the Cabinet held in 25th of July, 2000. Since Turkey needed to concentrate on a program of economic stability aiming to reduce inflation rates to reasonable figures, the government could not afford the estimated three to four billion US dollars needed for construction of the country's first nuclear power plant.

The Government also stated that;

— It was more preferable to build natural gas power plants in the short term, like other OECD countries.
— In 15-20 years period, if natural gas becomes scarce and less economical then it will be better to reconsider the nuclear power option.
— In order to be prepared for a natural gas crisis, construction of large numbers of NPP's required. Since the country's resources are limited, it is not possible to achieve such projects without external loans and such huge external loans might endanger economic programs.
— Therefore, continuation of hydro and natural gas projects and wait for the decrease in NPP costs and increase in their lifetimes would be better.
— During this period, there may rise an opportunity to utilize thorium as a nuclear fuel.
— It is not planned to cancel plans to build NPP's in the future.

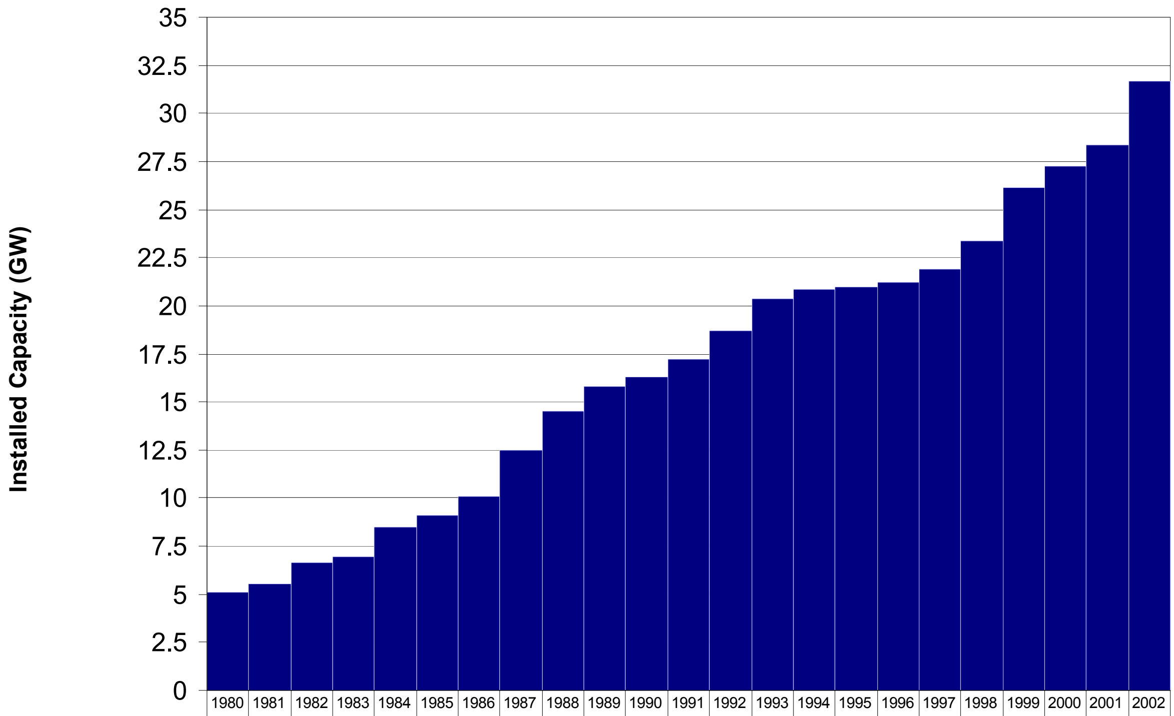
2. PRESENT SITUATION

2.1. Energy situation

Energy has been a priority government investment sector for some time and in 1986 received the second largest allocation of foreign financing among public-sector investments. Although limited, Turkey has some energy sources; coal, uranium, lignite, some oil and gas deposits, and considerable potential for hydroelectricity. In the year 2001, approximately 40% of primary energy consumption was met by petroleum, 30% by coal, and 20% by natural gas. Although, during the period of 1996-2000 the primary energy consumption rate has increased by 4.5% per year and reached to 81.2 Mtoe by the year 2000; because of economical crisis and negative economical growth it dropped to 77.0 Mtoe in the year 2001 [1]. The electricity consumption was increased about 7.5% per year, during the same period, and reached about 132 TW·h at the end of 2002. The installed capacity for electricity generation was about 31.6 GW_e by the end of April 2003. Annual development of electricity generation capacity is given in Figure 1 between 1980-2002. Electricity consumption per capita is about 1,955 kW·h by the year 2002 [2]. National electricity data is given in Table I. Turkey is not rich in energy resources and import dependency was about 72% by the year 2001 and will increase in time as the energy consumption increases. It is expected that the annual electricity demand rate will increase about 8% - 10% till 2010. The projection for electricity consumption reveals that about 290 TW·h will be consumed by the year of 2010 and the required installed capacity will be around 59 GW[3]. The share of fuels for installed electricity generation capacity and the share of electricity production is given in Figures 2 and 3 respectively for the year 2002. The CO_2 emission was about 210 Mton in 2000 and is expected to increase to 390 Mton in 2010. The CO_2 emission per capita is not so high compared to the world and OECD averages, i.e. 2.9 tons/capita in Turkey which is much less than those of world average (3.87 tons/capita) and OECD average (11 tons/capita).

Table I. Electricity data for 2002 [2]

Total Electricity Production	129,367 GW·h
Thermal	95,602 GW·h
Hydro	33,717 GW·h
Wind	48 GW·h
Total Electricity Consumption	132,519 GW·h
Per capita Consumption	1,955 kW·h



	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002
Installed Capacity(GW)	5.119	5.538	6.639	6.935	8.462	9.122	10.12	12.5	14.52	15.81	16.32	17.21	18.72	20.34	20.86	20.95	21.25	21.89	23.35	26.12	27.26	28.33	31.67
Increase (%)	0.00	7.56	16.58	4.28	18.04	7.24	9.82	19.05	13.95	8.15	3.12	5.18	8.05	7.97	2.50	0.45	1.39	2.93	6.26	10.59	4.20	3.77	10.53

FIG. 2. Annual development of Turkey's installed electricity generation capacity between 1980-2002 [4].

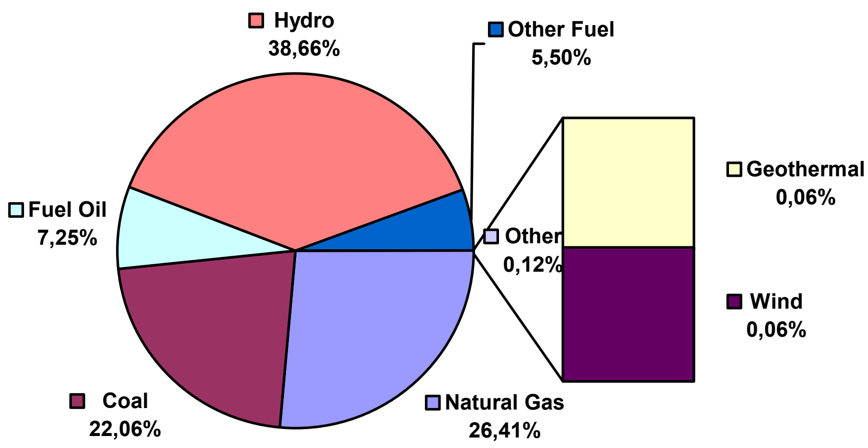


FIG. 3. Share of installed electricity generation capacity by source (2002) [2].

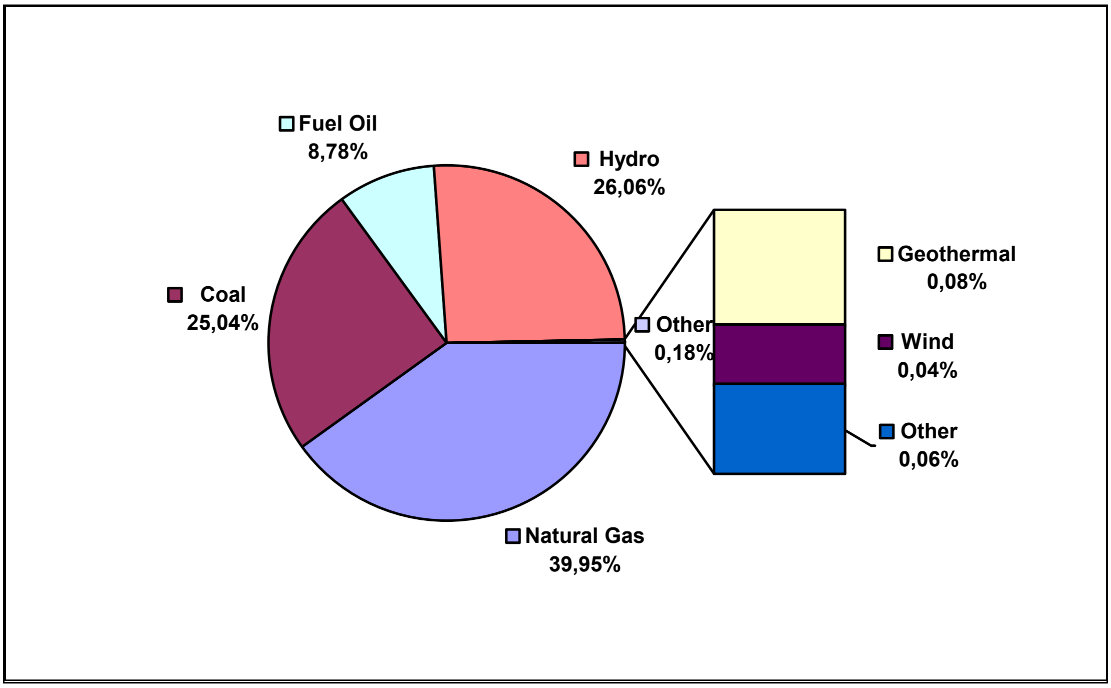


FIG. 3. Share of electricity production by source (2002) [2].

2.2. New electricity market law

The "Electricity Market Law" no: 4628, published in the Official Gazette dated 3 March 2001, is enacted to unbundle electricity market activities, enable progress into a liberalized electricity market and provide for fair and transparent market regulation.

The main objective of this law is to create a competitive electricity market with the great majority of the participants in this market being private companies and most of the assets used to supply electricity being privately owned. The role of the State would be greatly reduced. The law;

— creates a framework in which there will be privately owned generation companies, privately owned retail and wholesale companies, and Government owned transmission company. The private generation companies will sell electricity to the supplier companies with the transmission company transmitting the electricity. All participants must obtain license to perform activities in the market;
— allows larger electricity consumers (eligible consumer) to buy power from any source rather than just from the local distribution company. This is designed to introduce competition and to meet the requirement of the European Electricity Directive;
— establishes a Regulatory Body to issue licenses and oversee the market; especially the transmission and distribution charges and the tariffs for captive consumers;
— limits monopoly in the sector;
— allows the privatization of electricity assets according to the Privatization Law (Law no:4046);
— allows all those Transfer of Operating Rights (TOOR) which cannot be finalized by June 30, 2001 to lapse;
— limits Treasury Guarantees for new Build-Operate-Transfer (BOT) projects and Build-Operate-Own (BOO) projects to certain plants which have already been agreed between

Treasury, State Planning Organization (SPO) and Ministry of Energy and Natural Resources and even then the guarantee applies only if these plants can be in operation before the end of 2002;

Generation, transmission, distribution, wholesale, retail-sale and retailing services, import, export of electricity and the establishment of the Energy Market Regulatory Authority and rules and principles related to it's operations, is the subject of the law.

The Electricity Generation Company and private sector generation companies may sell electricity and/or capacity to customers in accordance with their licenses. The Electricity Generation Co. Inc. may build, lease and operate new generation facilities on behalf of the State where deemed necessary in accordance with the Board-approved generation capacity projection, taking into account private sector generation investments. The total market share of a private sector generation company, which it may acquire through the generation facilities it operates together with its affiliates, may not exceed twenty percent of the published figure for the total actual installed electricity generation capacity in Turkey in the preceding year.

The Turkish Electricity Transmission Co. Inc. is empowered to take over all transmission facilities owned by the public and plan the transmission investments for the proposed new transmission facilities and to build and operate these new transmission facilities. The Turkish Electricity Transmission Co. Inc. will perform international interconnection activities in line with the decision of the Ministry and will provide transmission and connection services to all system users including eligible consumers connected and/or to be connected to the transmission system, without discrimination, in accordance with provisions of grid code and transmission license.

Distribution companies in areas specified in their respective licenses shall conduct the electricity distribution activities.

The Turkish Electricity Trading and Contracting Co. Inc. and private sector wholesale companies shall conduct wholesale activities.

Retail sale companies shall conduct activities involving retail sale of electricity and/or capacity and of retail sale services. Retail sale companies are allowed to engage in retail sale or retail sale service activities without being subject to any regional limitation.

This law defines the framework, but requires a large number of secondary regulations and procedures to make it work properly.

3. FUTURE SCOPE

3.1. Energy policy

The Turkish energy policy is mainly concentrated on assurance of reliable, sufficient, economic and clean energy supply in time, and in a way to support and orientate the target growth and social developments. The energy planning studies shows that energy demand of the country will increase with development and industrialization. In order to meet the demand reliably, significant increase is expected both in energy production and in supply in coming years. Turkish environmental policy considers that energy policy should take into account environmental problems and a balance should be established between increases in energy demand, which is required for economic development, and environmental concerns. Some of the main criterions which are stated in the Seventh Five-Year Development Plan are given below:

— A dynamic and feasible master plan which accomplishes the optimum planning of resources in an economical and reliable way and which minimizes the environmental problems during the production and consumption of energy has been prepared.
— For Turkey, it is necessary to meet energy requirement with national resources as much as possible and to use new technologies, which eliminate the adverse effects of energy production on the environment.
— According to the development plans, “energy saving” is one of the basic principles.

3.2. Expectations from nuclear energy

Since future nuclear power program of Turkey is dependent on nuclear policy, Turkish Atomic Energy Authority (TAEK) has recently initiated a project to revise the nuclear policy of the country. This project includes various application sectors of nuclear energy, including nuclear power, and programs associated to each sector. One of the sectors that should be considered is the “Research and Development” which also includes innovative designs and small and medium sized reactors (SMRs). Cooperation with international/national groups on theoretical and experimental projects concerning SMRs and innovative technologies would lead to an increase of staff capabilities and experience on nuclear technology in Turkey. To achieve this goal, TAEK decided to participate in the “International Project on Innovative Nuclear Reactor Technologies and Fuel Cycles (INPRO)” and also became a member to the IAEA’s “Technical Working Group on Gas Cooled Reactors”.

The primary concern of the Turkish Atomic Energy Authority for innovative nuclear reactor technologies are in the areas of: (i) resources, demand and economics, (ii) safety, (iii) environment.

3.2.1. Resources, demand and economics

Turkey has some energy sources like coal, uranium, lignite, some oil and gas deposits, and considerable potential for hydroelectricity. The known uranium reserves in Turkey were reported as 9129 tons; however this amount of reserve needs further investigation with respect to grade and feasibility. The same holds true for thorium reserves, as fertile material, which was reported as 380,000 tons. It should be noted that Turkey should have a long-term policy for using own natural resources for generating nuclear energy. However, today the cost of uranium favors the utilization of import uranium fuel for short and medium-term nuclear fuel supply.

The projections of the Ministry of Energy and Natural Resources reveal the fact that installed capacity and generated energy will increase rapidly in coming 20 years. The installed capacity is estimated to be 46 GW with 290 TW·h energy productions in the year 2010 and these figures are estimated to become 88 GW and 547 TW·h by the year 2020. The additional installed capacity breakdown for the next 20 years reveals that gas power stations will dominate to the other type of technologies such as coal, hydro and nuclear. Gas used for electricity generation and heating is imported from various countries. Although the most important potential risks for gas import for generating electricity are stability of gas cost and political conflicts, lower electricity generation cost and shorter construction period favors the utilization of this fuel for electricity generation. The capital cost of a gas fired plant is about 400-600 $/kW, however the cost is about 1500-3500 $/kW for nuclear power plants. It is clear that two factors are to be considered for nuclear reactors to be able to compete with combined cycle gas stations:

— Capital and generation costs
— Construction period

If the capital cost would be around 1000 $/kW then the financial burden could be much less than current nuclear technologies which could ease launching nuclear power programs in especially developing countries. The main reason for postponing Akkuyu NPP project was the financial burden due to external credit needed for the project. The innovative nuclear reactor technologies should focus on decreasing capital cost, without compromising safety. In Turkey, after the new Electricity Market Law, the private sector will lead to new investments along with privatization of electricity generation plants. The range of capital cost per kW will play an important role for selection technology in future and reduction of capital cost of nuclear stations could favor the use of nuclear energy stations by private investors. Another option for private investors in Turkey could be the use of small sized nuclear generation stations. At this point distributed power concept can be considered. The construction period, on the other hand, is important in developing countries including Turkey since energy demand increases at higher rates (8-10% per year) and delays in construction periods sometimes could lead to undesired over capacity. In developing countries however planning of new capacities is very dynamic and can change in short-terms.

3.2.2. Safety

The improvement of safety of nuclear reactors is always in progress inline with technological improvements and lessons learned from various applications. Indeed, the accidents at the Three Mile Island and Chernobyl nuclear power plants have led to a momentum for improving safety technology and even the safety philosophy. We learned from both accidents that the *defense in depth* concept is important for safety and risk perception could be changed easily even after one serious disaster at a commercial reactor per 438 reactors in operation with more than 9000 reactor-years operating experience. Following items summarize the factors to be considered for innovative reactor technologies:

— Inherent safety philosophy should be applied to the design of innovative reactor technologies along with the requirement that each sequence of events leading to an accident condition should be well evaluated.
— In case core melt is not avoidable the core melt frequency should be 10^{-6} or less.
— The reactor should have a self-protection system against insertion of maximum possible reactivity.
— The licensibility of nuclear power plants could be important for developing countries since there is not so much experience. In that case, a licensed reference plant could be a solution for innovative designs.
— The NPP developers should take care of two main points: 1) to develop safe and economic design and 2) to provide conditions in which the public would accept this design as safe and economic.
— The use of passive mechanisms for heat transport system and/or safety systems is generally a better solution for simplifying designs that also decreases the cost however performance of these systems should be well demonstrated against accident conditions. The use of passive systems can also improve availability of those systems upon demand.

3.2.3. Environment

The position of TAEK is such that "*environment*" as a subject area plays an important role for the concept of "*sustainable development*". The share of fossil fuel for energy generation in the world is about 77% and this figure reaches to 90% in some countries. The situation is not different in Turkey and the share of fossil fuels is as high as 90% in primary energy consumption. The fossil fuels dominate in electricity generation as well, i.e. about 60%. From the greenhouse gas emissions point of view, Turkey is not that critical since the CO_2 emission per capita is as low as 3/4 of world average. However, in the year 2020, the current installed capacity will be tripled and greenhouse gas emissions will be increased accordingly. Hence clean technologies like nuclear and renewable sources will be unavoidable in developing countries like Turkey if greenhouse gas emissions should be stabilized at certain levels, as international agreements and protocols (like Kyoto Protocol) dictate. However, high level radioactive waste seems to be the most important drawback of nuclear technology as far as public acceptance is considered since the solution for commercial reactor waste disposal could not be demonstrated yet. The position of TAEK is that the technology for such geological disposal is available but it is to be demonstrated for public's acceptance. There may be two approaching points for this problem: 1) innovative nuclear reactor designs can be such that their high level wastes could be less than current technologies and 2) An innovative solution to deal with the high level waste may be developed such as transmutation of long lived wastes to short lived isotopes by use of accelerator driven systems.

4. INNOVATIVE TECHNOLOGIES AS A RESPONSE TO TURKISH REQUIREMENTS

World energy demand is increasingly getting dominated by developing countries like Turkey. Most of these countries need to improve their technological and human resources infrastructure and to get financial support for the introduction of nuclear technologies in power generation.

There are some factors negatively effecting the utilization of nuclear energy in developing countries. High capital costs of nuclear energy systems leads to heavy burdens unbearable by the fragile economies of developing countries. Long construction periods causes over capacity and under capacity problems due to dynamic nature of energy markets and the difficulties in long term planning. Competition of other energy sources, especially, natural gas, in terms of relatively shorter construction periods, lower capital and generation costs make it disadvantageous to build nuclear power plants in the short term. Chernobyl and TMI accidents and exploitation of sensational nature of nuclear energy by mass media have a strong negative effect over public acceptance.

In view of all these factors and national priorities, Turkish Atomic Energy Authority started a study to determine the criteria for innovative nuclear energy systems. Ultimate goal of this study is to find out the proper innovative technology fulfilling Turkish requirements for the future. Outcome of this study so far can be summarized as following criteria:

— Capital cost of the system should be less than 1000 $/kW and the system should be capable of producing energy less than 4 US cent/kWh.
— The construction period should be short enough to allow competition with alternative energy sources especially with combined cycle gas plants.
— Licensing of the plant should not be too complex and licensing period should be as short as possible.

- Safety of the innovative system should be enhanced by means of inherent safety features and simplified safety systems.
- Innovative concepts utilized in the system should be based upon proven technologies.
- When the energy system taken into consideration as a whole with all processes, components and material flows during their life cycle should have a better environmental performance than the current technologies.
- Overall innovative energy system should generate less waste by means of volume and toxicity than the current nuclear energy technologies.
- Overall energy system should utilize resources better than the current technologies.
- Innovative energy system should utilize indigenous resources, to the extend possible. Ease of technology transfer and contribution of domestic industry are important factors in order to decrease external dependency.
- Diversity of the applications is important, such as suitability to hydrogen production, desalination or process heat production.

Most of these requirements cannot be met by the current generation of nuclear reactors. There is a strong need for the development of innovative systems. If the studies in innovative reactor technologies end up with commercial/feasible designs covering economical and safety requirements then this kind of reactors can be a considerable option for energy production composition of developing countries. Otherwise, nuclear energy may encounter some difficulties to fulfill the requirements of developing countries dominating the world energy market day by day.

Nuclear energy has the potential to contribute to solving the problems posed by fossil fuels provided that it becomes economically competitive and publicly acceptable. It should also be noted that the governing factor for defining the level of development for a country is not only the energy consumption per capita but the technological infrastructure that will be handed over next generations.

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RENOVATION OF THE PNEUMATIC TRANSFER SYSTEM OF THE DALAT NUCLEAR RESEARCH REACTOR

TRINH DINH HAI, NGUYEN THANH CUONG, HUYNH TON NGHIEM, GUYEN NHI DIEN, PHAN QUOC MINH AND PHAM XUAN PHUONG

Nuclear Research Institute
Dalat, Viet Nam

Abstract. Since its improving from the previous system, the Pneumatic Transfer System (PTS) of the Dalat Nuclear Research Reactor (DNRR) has been playing an important role in the field of Instrumental Neutron Activation Analysis (INAA). However due to using discrete and low-level integrated electronic components, the system's technology was somewhat obsolete. After ten years of operation, the system's reliability was degraded because of ageing effects on equipment and electronic components. Furthermore, this system is unable to perform the cyclic INAA (CINAA) technique. Renovation of the PTS is imperative not only for increasing its reliability but also for expansion in functioning and convenience in operating. Following this spirit, the PTS has been renovated in the framework of the Vietnam Atomic Energy Committee (VAEC) Project, named "Renovation of the Pneumatic Transfer System of the Dalat Nuclear Research Reactor". Renovation and modernization tasks were implemented from March 2000 to February 2002. The main results are redesign and construction of the PTS's control system which based-on PC and construction of the measurement system on the spot based on gamma – ray spectrum system using semiconductor detector Ge for research on the CINAA technique with short – lived radioisotopes. The renovated Pneumatic Transfer System has been checked, commissioned by the VAEC and put into operation since February 2002. This paper describes the renovated PTS with emphasizing on its PC – based control system.

1. INTRODUCTION

The Dalat Nuclear Research Reactor (DNRR) was reconstructed during 1982 - 1984 period from the previous 250 kW TRIGA MARK II reactor installed in 1963. The reconstruction had been completed at the end of 1983. The reactor core, the control and instrumentation system, the primary and secondary cooling systems as well as other associated systems were newly designed and installed. The PTS was installed for two irradiation channels in reactor. The first channel, namely 13-2 channel is received a neutron flux about 10^{12} n cm^{-2} s^{-1} at the irradiation site and the second channel, namely thermal channel is received a neutron flux about 10^{10} n cm^{-2} s^{-1} at the irradiation site. The renovated reactor reached its criticality in November 1983 and obtained its nominal power of 500 kW in March 1984. Since then DNRR has been operated safely. It is mainly used for research, radioisotope production, neutron activation analysis and training.

In the field of neutron activation analysis, the INAA technique generally involves a neutron irradiation followed by a decay period of several minutes to many days. INAA can be classified into six categories depending on the half – live of the nuclide produced. These are [3]:

(i) Very short lived nuclides (half – live < 500ms),
(ii) Short lived nuclides (500ms – 100s),
(iii) Short to medium lived nuclides (100s – 10min),

(iv) Medium lived nuclides (10min – 15h),
(v) long lived nuclides (15h – 365d)),
(vi) Very long lived nuclides (1 – 5 year).

In researches on short – lived nuclides, CINAA is employed whereby a sample is irradiated for a short time, quickly transferred to a detector and counted for a short time. This process of irradiation – transfer – counting is repeated for an optimum number of cycles.

The PTS was redesigned and constructed to meet requirements for reliability, safety in operation and for researches on short – lived nuclides. A detailed description is given below.

2. DESCRIPTION OF THE RENOVATED PNEUMATIC TRANSFER SYSTEM (RPTS)

The RPTS consists of four main parts:

— Operating Mechanical System,
— PC – based Control system,
— Measurement System on the spot based on gamma – ray spectrum system,
— Software program for the PTS's control system and for start-up the Measurement System on the spot.

2.1. Operating mechanical system (OPS)

The OPS, shown schematically in Figure 1, consists of the following major components:

— A specimen capsule
— A blower – and – filter assembly
— Two specimen loading boxes
— A measuring – taking specimen box
— Four solenoid – operated valves
— Two channel – selected valves
— Four sensor blocks

The specimen capsule, or "rabbit", is made of polyethylene. The effective available space inside the capsule which used for the 13-2 channel is 14.2mm in diameter by 100mm in length, giving a usable volume of 15.9cm^3. The effective available space inside the capsule which used for the thermal channel is 10mm in diameter by 50mm in length, giving a usable volume of 3.9cm^3. *The blower – and – filter assembly* installed on a wall-mounted steel angle support, this assembly consists of a blower, a manifold, plenum chambers and a filter. The blower exhausts the system air into a vent pipe that discharges outside the building. *The specimen capsule loading boxes* for both channels allow loading in maximal 5 rabbits all at once. *Four solenoid – operated valves* control the air flow. *Two channel – selected valves* will allow the PTS to be operated with the thermal channel or the 13-2 channel. If the 13-2 channel is selected, T1 is closed and T2 opened. If the thermal channel is selected, T1 is opened and T2 is closed. *The measuring – taking specimen capsule box* has a 70cm by 70cm face area and is 60cm deep, is covered by a class of pure lead with thickness of 10cm outside and by three classes (aluminum, copper and mica with thickness for each class is 1cm) inside. It was designed with consideration to the type and level of radiation in the surrounding areas so that the gamma ray spectrum of a sample is not influenced by the background radiation. *Four sensor blocks*, in which D1, D2 used for the thermal channel and D3, D4 used for the

13-2 channel, will allow to trace out of rabbit's position before into or out from the core and measuring – taking box.

The system is controlled from the control cabin and may be operated either manually or automatically. The system operates on a pressure differential, drawing the specimen capsule into and out of the core by vacuum. Thus, the system is always under negative pressure so that any leakage is always into tubing system. All the air from the PTS is passed through a filter before it is discharged.

In order to reduce dead time of the measurement system on the spot, the detector block is putted on a "rail - car" which enables to adjust the distance between its and the measurement site by a two way electromotor.

For influential reduction of the radiation air, a blower is installed inside *the measuring – taking specimen capsule box.* The blower exhausts the radiation air into a vent pipe that connected directly to the main ventilation of the reactor hall.

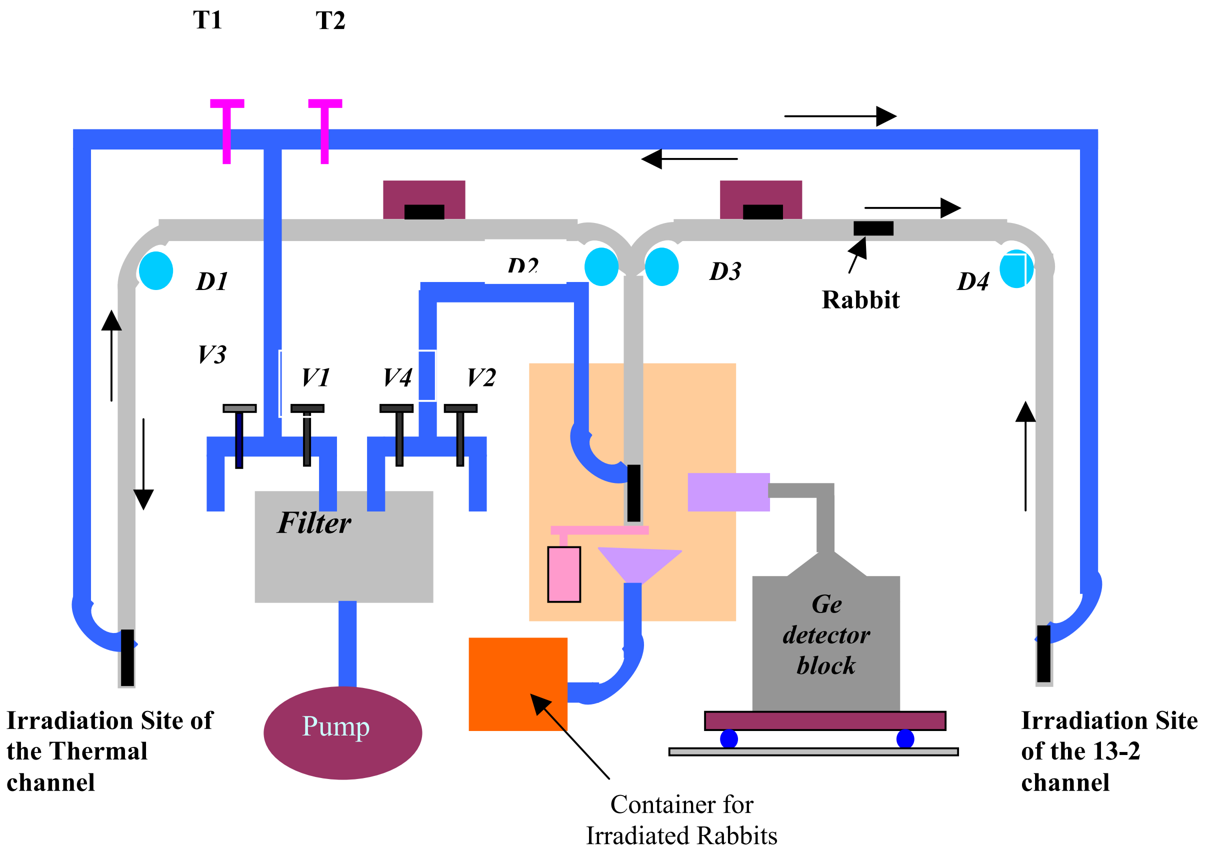

FIG. 1. The block diagram of the renovated pneumatic transfer system.

2.2. PC – based control system

The control system was designed and constructed in new completely. This system may be operated in automatical mode or manual mode without PC support. In Figure 2 the block diagram of the PC – based control system is shown.

The hardware of the system consists of the following major parts:

(i) *The Central Unit*: this unit contains a power supply board for the RPTS, the functional electronic modules and an indication block for display of irradiation time of rabbit and necessary information in the manual mode.
(ii) *The specimen loading unit*: contains a loading box, a two-way electromotor and a control board. There are two specimen loading units, one for the RPTS on the thermal channel and other for the RPTS on 13-2 channel.
(iii) *The specimen taking unit*: located inside of the measuring – taking specimen box. This unit consists of a specimen taking control part used two – way electromotor and a control board.
(iv) *The sensor unit:* this unit was designed to detect rabbit before moving into or out from core and measuring - taking box. Its reliability is very important for operation of the RPTS. In our case, we used infra – red principle for the sensor unit. Four sensor units were installed.
(v) *PC Add – on Interfacing card:* this card is located in the extension slot of PC.
(vi) *PC Pentium II-* 266: this computer is used for both the PTS's control system and the measurement system on the spot. A multi – channel analyzer (MCA) add –on card is also located in the extension slot of this PC.

2.3. Measurement system on the spot based on gamma – ray spectrum system

The measurement system consists of three parts: a semiconductor detector Ge with resolution of 2.5 keV at peak 1332keV of Co^{60}, a NIM crate with the functional boards such as preamplifier card, amplifier card, high voltage card and the central unit with a PC and ADC/MCA add-on card. The block diagram of the measurement system is shown in Figure 3.

2.4. Software program

The software program of the RPTS was written in Turbo C and Assembler languages. Its flowchart is shown in Figure 4. The software is user – friendly and the most of information on the PTS such as transfer time, irradiation time, measurement time, number of cycles, position of rabbit, etc., are displayed on the screen under digital or graphic type and saved in hard disk for the task of operation management.

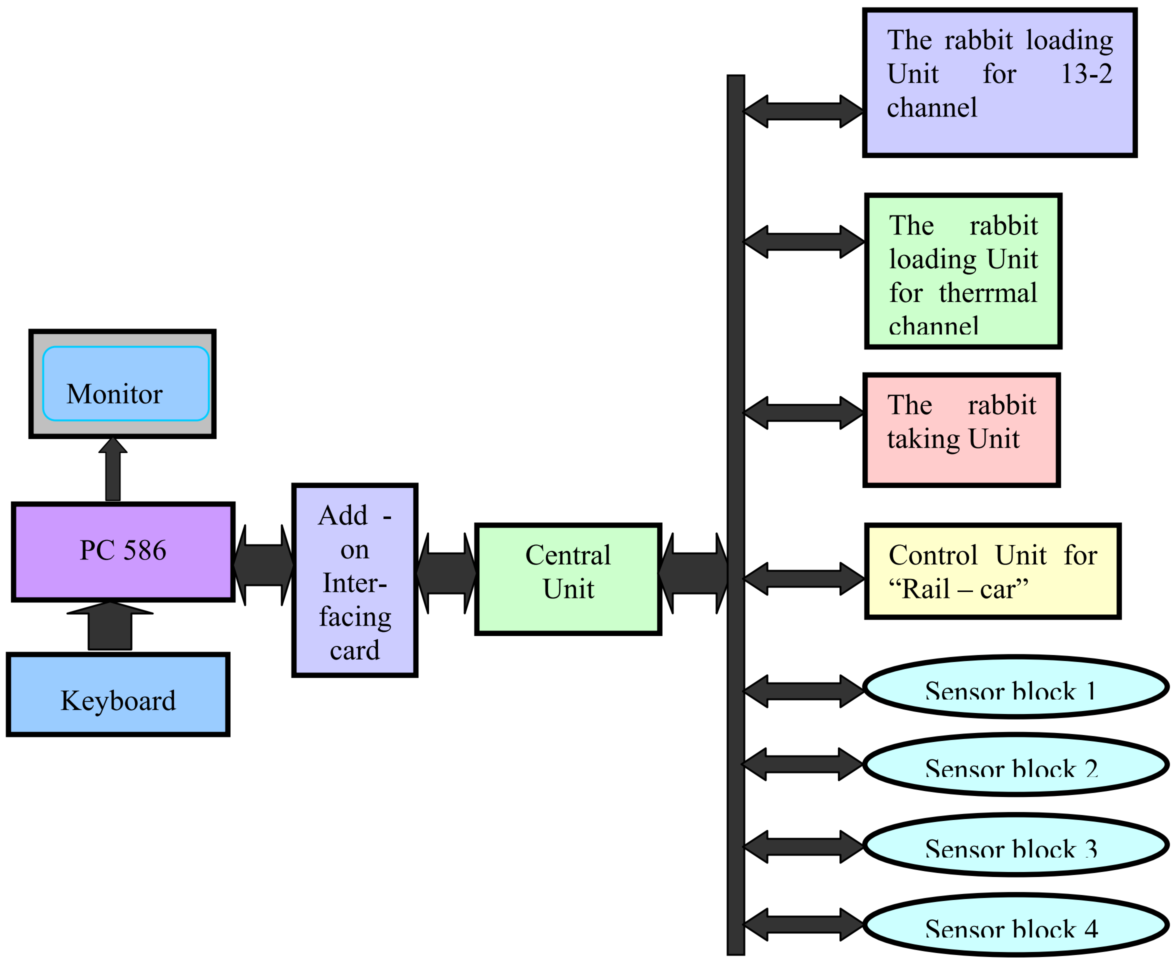

FIG. 2. the block diagram of the pc – based control system of the RPTS.

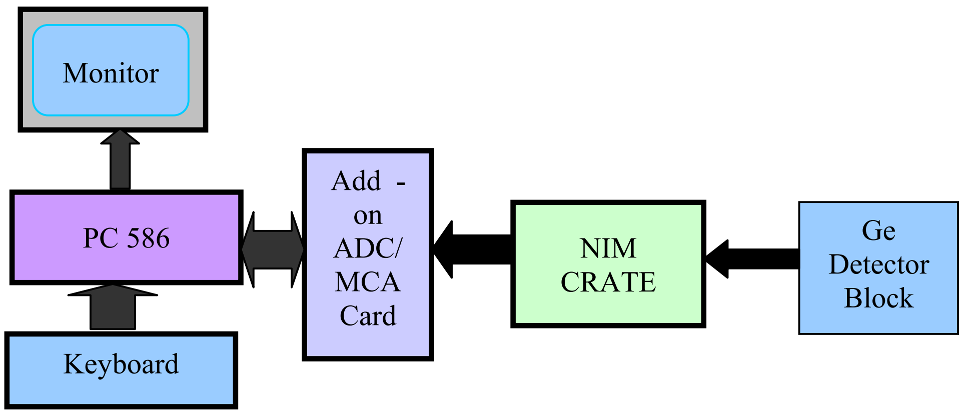

FIG. 3. The block diagram of the pc – based measuremenrt system on the spot.

3. MAIN TECHNOLOGICAL SPECIFICATIONS OF THE RPTS

— Simple and safety in operation with two mode: Manually and automatically
— The length of the rabbit pipelines from the measuring – taking specimen capsule box to the irradiation site are about 15m with the 13-2 channel and 4m with the thermal channel.
— The transfer time of rabbit from the specimen capsule loading box to the irradiation site and return is 2s – 2.2s with the 13-2 channel and 1s – 1.1s with the thermal channel.
— The system enables the user to irradiate rabbits one by one continuously with predetermined length of irradiation times for every one of them.
— Permitting to irradiate a rabbit with cycle of irradiation – measuring – irradiation with a predetermined number of cycles.
— The process of irradiation – transfer – measuring – taking with one or more of irradiation capsules is carried– out automatically.
— The distance between detector block and measurement site, which contained irradiated capsule, enables to chance from 2cm to 20cm on the horizontal direction by a “rail – car” with the moving speed is 2.8cm/min. This “rail – car” is controlled by the user from control cabin.
— The control and measurement center is installed in a cabin which is located in the reactor hall.

4. CONCLUSION

The renovation and modernization of the PTS were carried – out by staff members of the Nuclear Research Institute. The RPTS was checked, commissioned by the VAEC and put into operation since February 2002. Until now, the system is working with high reliability, convenience for users and easy for repairing and maintenance works. In the implement process, a lot of technological problems such as how to detect exactly rabbit’s position before into or out from the core and measuring – taking box, how to design and construction of the rabbit pipeline with short length and suitable style for rabbit’s moving smoothly, how to brake softly the irradiation capsule in to measurement site, etc. are solved strictly. The radiation background in the measuring – taking box is increased by the radiation dusts discharged from the rabbit pipeline during operation process. In order to reduce their influence, the rabbit pipeline must be cleaned in periodic.

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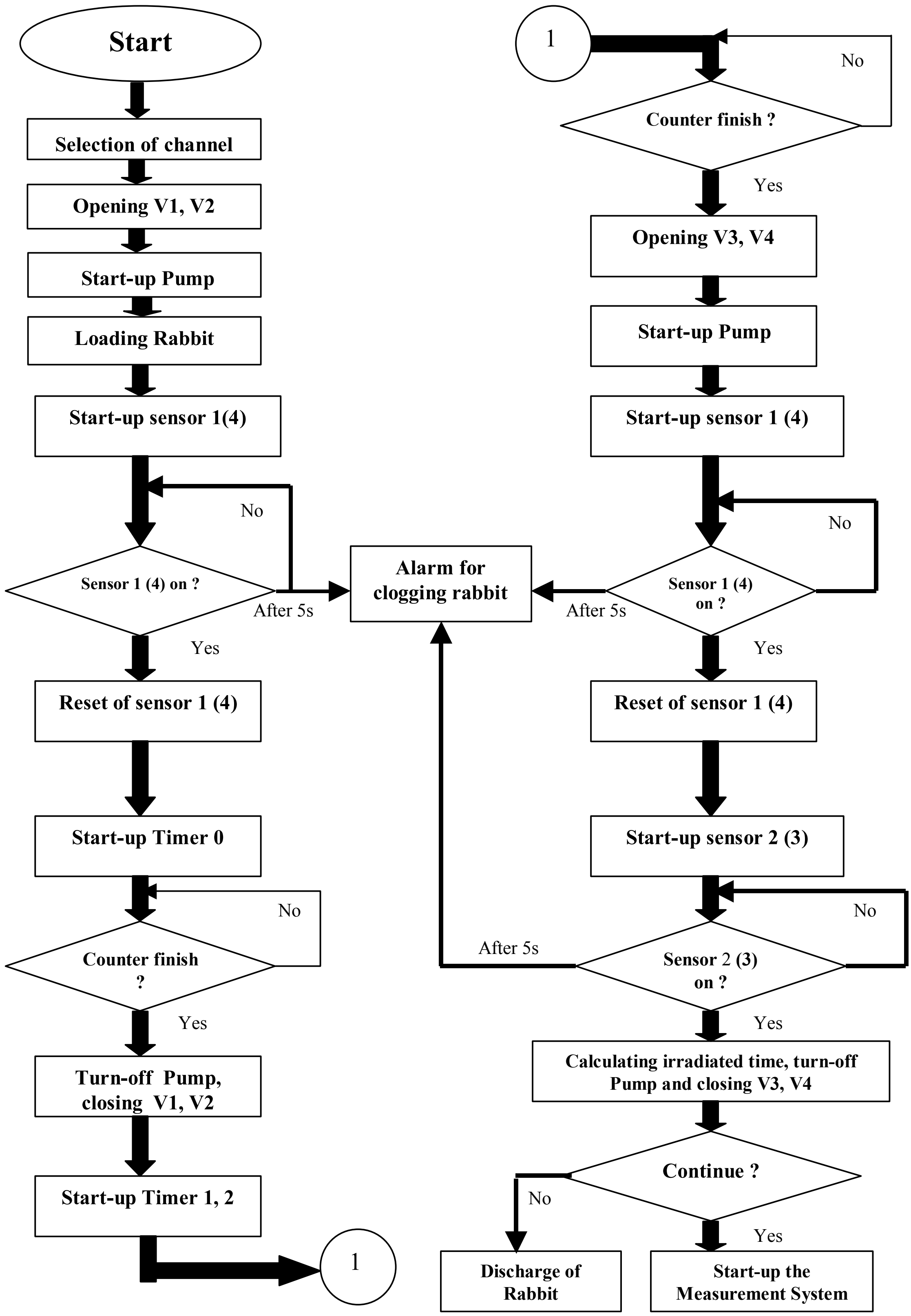


FIG. 4. The flowchat of the software program.

FUTURE OF LARGE SCALE NUCLEAR POWER DEVELOPMENT ON THE BASIS OF HOLISTIC SYSTEM OF NUCLEAR TECHNOLOGIES

P. ALEKSEEV, YU. CHERNILIN, N. KUKHARKIN, N. PONOMAREV-STEPNOY

Russian Research Centre "Kurchatov Institute"
Moscow, Russian Federation

Abstract. The decisive step in the development of nuclear power must be its recognition by the international community as an energy technology which, with practically unlimited resources and combined with renewable energy sources, can solve the fundamental problem of sustainable power development and, at the same time, reduce environmental pollution, including greenhouse gas emissions.

1. INTRODUCTORY REMARKS

Almost 20 years of stagnation in the nuclear power sector have led to a situation where a number of countries which have actively participated in its development and have achieved notable successes have now started to talk about cutting back their nuclear programmes. These are mainly Western European countries (Germany, Belgium, etc.) and their attitude has had a negative impact on the nuclear power programmes of some countries which were planning to introduce nuclear power, or to speed up the development of programmes they had already embarked upon. There are many explanations for this negative attitude towards the development of nuclear power, but all the arguments against nuclear power have no real scientific and technical, or strategic basis.

At the same time, greater attention is being given to (at present, mainly by that part of the scientific community working on power-related issues), and a body of opinion is gathering on the need for large-scale development of nuclear power in the relatively near future, a point which draws ever closer. However, the attitude of the public towards nuclear power remains uncertain.

The decisive step in the development of nuclear power must be its recognition by the international community as an energy technology which, with practically unlimited resources and combined with renewable energy sources, can solve the fundamental problem of sustainable power development and, at the same time, reduce environmental pollution, including greenhouse gas emissions.

One highly important factor giving us serious grounds to assume that the problems currently afflicting nuclear power can be solved successfully is the latter's negentropic potential, i.e. the enormous energy potential of fission (more than four order of magnitude greater than the energy of normal fuels) which can help solve many current energy generation problems (e.g. radioactive waste management, etc.) and eliminate nuclear power's present inherent shortcomings.

It is becoming ever more obvious that the future development of nuclear power is linked to the need for its large-scale use, which will require solving a large number of new problems (of a fundamental and complex nature, and requiring a systematic approach) caused by the problem of scale. Moreover, the large-scale development of nuclear power will necessitate a major restructuring of the power generation sector, which is a highly inertial and capital-intensive branch of the economy, and this too is a complex technical and economic problem.

2. SOME CONSIDERATIONS ON LONG-TERM NUCLEAR POWER DEVELOPMENT (GENERAL AND CONSERVATIVE VIEW)

Nuclear power will play an important role in the world future sustainable energy mix. But today such role is not assured. To understand nuclear power sustainability it is necessary to consider three major aspects:

— availability of nuclear fuel;
— compatibility of nuclear power with respect for the environment;
— economic competitiveness of nuclear power in the long-term.

2.1. Availability of nuclear fuel

Today known resources of uranium is around 3.8 million tones and 11 million tones of additional speculative (undiscovered) resources. Thermal reactors use only 1-2% of energy available from natural uranium. It means that from energy resources point of view today's nuclear power technology will not be sustainable options.

Breeder reactors could overcome uranium deficiency in two ways: by recycling plutonium produced in thermal reactors, and by converting fertile uranium (U-238) into fissile plutonium. Together these improvements could increase the energy extracted from natural uranium to 75% or higher. The reserves of uranium mentioned above could be extended by 40 times (a breeding fuel cycle based on thorium-uranium is also possible). In this case amount of fuel does sound really sustainable for a long time.

Fast or breeder reactors have not yet been verified by actual operation to the same extent as thermal reactors, although there is some experience of operation for fast reactors with sodium coolant and uranium fuel. A few demonstrated plants have been built and operated, including Russia's Beloyarsk plant BN-600 and BN-350 (now in Kazachstan), France's Phenix and Superphenix plants, Japan's Monju and some others. All breeder plants have proven to be more expensive compared to conventional thermal reactors. The technical potential of breeder reactors has been partially demonstrated, but results from the first few plants are not sufficient to define the precise features and costs of technically mature plants.

Notwithstanding the lack of experience, experts agree that fast reactors have a large safety potential. The reasons are: the possibility of the minimisation of reactivity effects up to the level lower than the share of delayed neutrons; the absence of high pressure in the first circuit; large margins for the boiling temperature, high degree of coolant natural circulation. In case of using Pb-Bi or even Pb as coolant instead of sodium, there is an additional advantage of the absence of latent chemical energy that would deteriorate the consequences of accidents with the loss of integrity of the first circuit.

Extending the nuclear energy supply via breeder reactors will require a fuel cycle with reprocessing. The essential elements of potential fuel cycles, mainly using the uranium-plutonium route have been demonstrated and do not pose any major technical problems. They do not appear to have any major economic problems, although the cost of implementing adequate safety measures for long term fuel cycle operation and waste disposal may not be fully known at present.

Conclusion on fuel availability: natural resources of energy supply for nuclear plants using current technology are limited. To extend this supply to a period on the order of centuries, breeder reactors and recycling of fissile material would be required. The technical ability of this approach to provide a very long term source of energy is clear, but its comparative economics are not.

2.2. Compatibility of nuclear power with environment

During normal operation of a nuclear power plant all potentially harmful radioactivity is controlled and kept within specifically identified areas of the plant. This protects first and foremost plant personnel, but necessarily protects the environment out side the immediate plant structures. Over time states authorities have instituted regulations on the control of radioactivity in power plants and these have proven to be effective in protecting plant staff, the public and the environment.

After the Three Mile Island and Chernobyl accidents it was clear that potential dangerous from nuclear power plants are accidental radiation releases. In the 1980s nuclear regulator bodies multiplied safety regulations, imposed many new design requirements, and constrained operating practices. The cost of plant systems, operation and staffing increased substantially. Improvement to nuclear power plant designs and operation must be a part of any view of nuclear power sustained in the long term. All the new reactors designs rely upon simplification of systems, passive safety systems requiring (in some cases) no power or operator intervention, and standardisation of design. The latter reduces the potential for improper field construction to introduce safety risks.

The experience increasingly proves that commercial nuclear power plants and fuel cycle facilities can be operated without serious accidents. Safe operation should become a smaller obstacle to the sustainability of nuclear power.

The disposal of high level radioactive wastes is the single important issue calling into question the sustainability of nuclear power. The disposal of radioactive waste has, since the beginning of nuclear power generation, been based on the principle of protecting human health and the environment. It has been manifestly clear that a primary goal must be to minimise the chance that the waste will come into contact with humans in the future. There is broad consensus on the merits of geological disposal of long lived radioactive wastes in deep and stable geological formations in order to do just that.

At present a long-term solution of the problem of handling nuclear wastes is reached nowhere. In most countries the wastes are piled expecting a long-term solution. Reprocessing of nuclear wastes applied in some countries does not change the overall picture much. In addition, the modern process of reprocessing the spent nuclear fuel that is based on aqueous methods (the PUREX process) has essential drawback, mainly due to the necessity of a long time before reprocessing, due to difficulties in returning minor actinides (Am, Cm) to the fuel cycle, due to large volumes of reagents and generated radioactive solutions that have to be discharged into the environment (discharge to lakes or oceans with therefore their radioactive contamination). Today some advanced concepts of radioactive waste handling under consideration. One of them is mentioned below. The first element of the system of handling of radioactive wastes is non-aqueous reprocessing of nuclear materials with the use of the fluoride gas, extraction, thermal diffusion and electro-chemical methods. This allows to divide spent fuel into three fractions:

— the fuel fit for reuse in power reactors;
— short-lived isotopes;
— long-lived isotopes.

There are the following advantages of non-aqueous chemical and pyro-chemical reprocessing as compared with traditional aqueous methods:

— compactness due to the absence of the moderator;
— simplicity of the design and equipment;
— ability to operate in case of short intermediate storage;
— small fuel amounts outside the reactor;
— absence of radioactive pollution of solutions;
— minimum requirements for the packing and storage of the materials.

This favourably distinguish this technology from the PUREX process used today, which assumes large volumes of reagents and, consequently, of wastes, and also has the following drawbacks:

— high concentration of a-active nuclides (^{238}Pu, ^{244}Cm) that results in quick deterioration of the quality of solutions. The level of cc-activity of the fuel after reprocessing is seven-fold higher than for uranium fuel of the same burnup;
— ^{14}C requires special attention: although this is a product of activation of ^{14}N traces in he fuel and construction elements, but its release into the environment is rather difficult to limit and it gets quickly involved into biological cycles;
— repeated fuel recirculation results in further complication of fuel handling, which necessitates a fully remote process of fuel fabrication;
— a long period of fuel recycle (4 years of irradiation in the reactor, 5-7 years for intermediate storage and cooling of the fuel, 1 year for fuel reprocessing, 1 year for fuel fabrication);
— some long-lived nuclides (Am, Cm) are constantly accumulated in successive cycles of fuel reprocessing.

The main drawback of non-aqueous reprocessing is that this is a new technology that has not yet received the same amount of attention and funding as the PUREX process. Accordingly, a rather serious effort is needed in order to demonstrate the feasibility of proposed methods and adequately substantiate the choice of materials, technical solutions etc.

2.3. Economic and/or competitiveness of nuclear power

The place of a new technology in the structure of the energy sector is determined by economic criteria. In this respect, one should note a complex situation with nuclear energy: high capital requirements for NPPs and for the fuel cycle create objective difficulties for the "start-up" of NP. For a state or, even more so, a private company to invest in such capital intensive projects, an assurance for a long-term is necessary. It is for this reason that the development of NP necessitates some state participation: the creation of a favourable national climate, a well-thought legislative and licensing basis (in order to avoid sudden and often changes of the approach to NPPs), existence of a development programme.

Assuming a sustainable energy supply to be the goal the availability of nuclear power economic sustainability must be with reference to other energy sources. In the long term, various factors affecting the relative costs of competing electricity generation technologies can change. An appraisal of long term economic sustainability cannot be based on conditions in today's energy markets or today's technology. The long term economic viability of nuclear power is critically dependent on fossil fuel prices, the value attached to reducing gaseous emissions from fossil power plants, and technological developments in nuclear and competing technologies. The level of safety demanded also influences nuclear power's cost. Emissions limits at the level of individual producers such as power plants would implicitly place values on carbon dioxide emissions. Market-based policies would make the value explicit. Power generation would be strongly affected by limits on carbon emissions or an explicit carbon value.

The relative competitiveness of different generating sources would be changed. The costs of nuclear energy would remain unaffected, but coal, oil and gas-fired generation would become more expensive. Restrictions on carbon dioxide emissions in the power sector, either implicit or market-based, could place coal-fired power generation at a significant cost disadvantage, while favouring gas-fired power. A carbon dioxide constraint could be an important potential contributor to nuclear power's economic sustainability in the long term, regardless of the ultimate resources of fossil energy available.

2.4. Other issues

Non-proliferation and public acceptance are issues that are certainly central to nuclear power's future in the near term. However, they seem less important when considering its long term fate. They are issues that will either allow nuclear power to proceed or will lead to its demise. That they will not ultimately block nuclear power's development.

The prominence of public acceptance as a challenge to nuclear power in the near term may be related to the present economics of energy supply. An important factor in the 1present situation is the relative abundance of energy at reasonable prices. If the future energy supply situation included substantially increased energy prices due to development of a carbon value, or simple growth in overall energy demand, public attitudes towards nuclear power could become more favourable as the balancing of nuclear's risks and rewards would shift.

2.5. Some observations

Nuclear power has the potential to be a sustainable energy source. There are no fundamental energy supply, environmental or economic issues which could exclude today to consider the nuclear power as a sustainable option for the future energy mix. But there are a number of serious challenges which nuclear power must face before it can play long term role.

The nuclear industry should develop appropriative long term technological solutions for full use of uranium.

As it was already mentioned extending the nuclear fuel supply via fast breeder reactors requires also a closed nuclear fuel cycle (with reprocessing). The society will comprehend its necessity only after the solution of the problems of the fuel cycle related to the selection of an acceptable technology of handling radioactive wastes. One can say that it is the acceptable means of liquidation and disposal of radioactive wastes that will determine whether nuclear power will exist or not.

Nuclear power must prove the competitiveness with other energy sources. It could be stated that the evaluation of nuclear power economic sustainability must be with reference to other energy sources.

The balance between safety and cost has not yet found its equilibrium Safety regulations should be based upon an appraisal of potential health and environmental impacts and the cost of minimising them. Safer at any price means never. Main problems listed above might be considered as a major tasks for development of special long-term programmes.

3. INPRO - STARTING TO MOVE IN THE RIGHT DIRECTION

In the nuclear power field, the IAEA has traditionally engaged in studies of a general nature focused on strategic problems, or on discrete, highly important issues (safety, economics, safeguards, etc.).

Regional studies are also organised, where the Agency sometimes acts as the initiator, but usually as the co-ordinator. The Agency takes a significant interest in specific, pressing, scientific and technical problems (of a regional and/or general nature). Experts from various countries deliberate upon these problems and the results, or conclusions and recommendations, are published in the form of reports (TECDOCs). Projects on the development of specific technologies and facilities, etc., can only be undertaken on the initiative of interested countries, and when practically all the funding for that initiative (project) comes from participating States. The Agency acts as co-ordinator and/or provides services, including - for example - organising additional independent expert assessment where this is required, holding conferences or meetings on research results, etc. The IAEA is neither financially nor politically able to implement a specific proposal (project). Problems of “duplication of effort” and competition with commercial companies, etc., arise.

The aims of INPRO, formulated when the project was set up, may be summarised in the following general terms:

— To support the contribution of nuclear power to meeting the energy needs of the 21st century on a sustainable basis;
— To bring together interested participating States, both those which develop technologies and those which use them, for the purpose of a joint review of the international and national efforts required to achieve the desired innovations in reactors and nuclear fuel cycles using reliable and competitive technology, based - as far as is achievable - on systems with inherent safety characteristics which also minimise the risk of proliferation and negative effects on the environment;
— To create a process involving all interested parties which will influence the activities of existing institutes and ongoing initiatives of national and international organisations, simultaneously utilising the results of these activities and adding to them.

Summarising the above:

The aim of INPRO is: To promote the economic and safe use of nuclear technologies with a view to meeting mankind’s global energy needs in the 21st century in a sustainable manner, while minimising the risk of nuclear weapons proliferation and the impact on the environment.

The general goals of INPRO are:

— To convince society of nuclear power's role in the 21st century;
— To bring together producers and consumers of nuclear technologies in joint activities at national and international level;
— To establish a mechanism for interaction between all participants and organise a process for the development of innovative nuclear technologies.

The work done under the project generally reflects these aims and goals, and the number of countries which have shown an interest and are participating in it in various forms is increasing.

However, there are real difficulties. They include the following:

— The major developed countries with large-scale national nuclear power programmes (USA, France, Japan, the United Kingdom) have not joined the project and are only present as observers in the Steering Committee. These countries, together with some others, have launched the Generation IV project which, in fact, is a grouping of nuclear technology producer countries that, in terms of its aims and on a scientific and technical level, coincides to a large extent with the aims and goals of INPRO. However, work under this project has mainly concentrated on reactor technologies which are actually a development of existing technologies;
— Two major countries in terms of their population (India and China), which are developing economically at a fast pace and which are clearly aiming at large-scale development of nuclear power, are participating in the project, but their level of activity does not correspond to their aims and potential and does not contribute to the strengthening and development of the project;
— To date, the INPRO project has been and is being financed only by Russia. This does not contribute to its strengthening and development. The Director General of the IAEA has proposed that the project be funded from the Regular Budget as of 2004 to the level of $600 000 per year. However, this proposal must still go through a number of review stages, including in the Board of Governors, and be approved at the IAEA General Conference.

The main activity under the project during this phase was the elaboration of a methodology for assessing innovative nuclear technologies with a view to determining their prospects, including **user requirements**, and **criteria** for establishing the level of compliance with those requirements, with the assistance of relevant **indicators**. In addition, the report and working materials include a chapter on the prospects and potential of nuclear power in the 21st century in the context of sustainable development requirements and adequate energy supplies for basic needs.

The various topics mentioned are currently at different stages of readiness. The sections on prospects and on user requirements for safety of nuclear reactors and fuel cycle facilities are most mature; those on economics and "related issues" (infrastructure, nuclear legislation, public acceptance, etc.) and proliferation resistance are least mature.

In most cases, a lot of work remains to be done on the details of the user requirements, formulation of criteria and selection of indicator values. At this stage, it is planned that work will proceed "from top to bottom" (from higher-level requirements to individual requirements) and vice-versa, with the help of experts from Member States working in situ to the specifications of the so-called Work Packages and Case Studies, proposals for which have been submitted to the Steering Committee. This work is in progress. The final report on

Phase 1A will be submitted to the Steering Committee in May and will be presented at the international conference on advanced nuclear technologies to be held by the IAEA at the end of June 2003.

These deadlines are rather tight, and intensive efforts will be needed from all experts working permanently on the project or assigned to it, as well as from the staff of the IAEA Secretariat. The details of the programme for continuation of work under the project will depend on Member States' assessment of the results of the first phase.

The following may be viewed as the main positive results of the first phase:

— The project has attracted several countries at very varying stages of development as regards their national nuclear power programmes, ranging from fairly wide commercial use (Canada, Switzerland, European Commission, Spain and South Korea) to countries taking their first steps (Argentina and Brazil) and making preliminary plans (Turkey, Chile and Croatia). Recently, Pakistan, Australia, South Africa, Rumania and the Czech Republic have expressed a desire to join INPRO. Belarus, Armenia and Ukraine are still considering whether to join INPRO, and Armenia and Belarus have declared their support for the project;
— The project is focusing on an area of nuclear science which, in recent years, has fallen into oblivion world-wide: the detailed and comprehensive study of the large-scale development of nuclear power, taking into account the potential role of innovative technologies. Studies of this type are being conducted intensively and co-ordinated globally for all other energy technologies which are either current competitors of nuclear power or will be competitors in the near future (organic fossil fuel, solar, wind, biomass);
— The project participants have been very appreciative of the research done by Russian experts on INPRO problems and on development paths for large-scale nuclear power.
— The Generation IV participants are beginning to recognise the significance of the project, as was demonstrated by the progress and results of the discussions at the last meeting of the Steering Committee where the observer countries participating in Generation IV were very active.

4. RECOMMENDATIONS FOR FURTHER WORK

Pursuant to the above, we may formulate the following recommendations for the near future:

— Work on the INPRO project should be supported, and a clearer national Russian position should be defined on the main areas of activity and explained to our experts in particular;
— In the short term, ways and means must be found to draw China and India's attention to the importance and potential of the INPRO research and the desirability of their playing a bigger role in the project;
— A wide range of experts from Russian institutes should be encouraged to participate actively in the completion of the work on Phase 1A, especially as regards the elaboration of the criteria and indicators for the INPRO methodology and their application in examples related to innovative nuclear technologies under development in this country;
— Some attempt should possibly be made at international co-operation with the developed countries to study the possibility of co-ordinating the work of INPRO and Generation IV.

For the first time in the last fifteen years, the Agency has been entrusted with a task which relates to the promotion of an activity fundamental to its role - support for the development of nuclear energy. In recent years, the IAEA's functions have been almost exclusively protective and verificatory: non-proliferation of nuclear weapons and safeguards, nuclear safety, radiation protection, etc. Though these tasks are clearly important, evaluation of the Agency's programmes and action related to the role of INPRO are not satisfactory at present. At the initial stage this is understandable, but this situation must change radically in the very near future.

At the level of general and/or long-term strategy, the INPRO project must be viewed as a broad-based programme of fundamental and applied research into problems relating to the large-scale development of nuclear power. Large-scale nuclear power means not only covering a significant proportion of the fuel and energy balance with nuclear power, but also its acceptability from all points of view, i.e. not only safe and economic nuclear power now, but also meeting society's requirements for the long-term future. The availability of enormous negentropic fission energy reserves gives us room to hope that all the specific problems of nuclear power which currently rouse public concern can be solved.

The symposium planned for June 2003 on innovative technologies which are a main component of the INPRO project should be used as a forum for discussing and shaping the INPRO programme.

5. GENERAL CONSIDERATIONS REGARDING A LONG-TERM (OR STRATEGIC) INPRO PROGRAMME

The need to speed up the solution of problems relating to large-scale development of nuclear power is determined by the following factors:

— Reduction in the transition times to new types of fuel: wood → coal → oil → gas → and now nuclear fuel. The introduction of coal in the power sector took a century, but the introduction of gas decades;
— Increase in the scale of fuel use to meet the growing needs of society for energy owing to the rise in global population numbers and increasing needs, and hence the need for shorter transition times to a new fuel structure;
— Need to create a man-made fuel cycle without use of natural ecological resources (air, water) by closing the fuel cycle;
— Simultaneous creation of all the entirely innovative infrastructure required and gradual restructuring of the existing fuel-energy infrastructure.

Thus we need to start now on research into the key problems of setting up large-scale nuclear power, creating and demonstrating new technologies, solving radioactive waste management problems, developing new approaches to safety problems, non-proliferation of nuclear weapons, etc.

The development of a long-term INPRO programme lies within the scope of the Agency's existing programme. Apart from the Department of Nuclear Energy, the setting up of such a programme would draw in (or take account of) the existing programmes of other Departments (Safety and Security, Safeguards, Research and Isotopes). Indeed, in time this programme would become a general programme of the Agency, signalling the start of a new stage of the IAEA's activities directed towards the development of large-scale nuclear power with all the consequences that brings with it.

SYSTEM OF SMALL NUCLEAR POWER STATIONS WITH AN APPROPRIATE INFRASTRUCTURE

P.N. ALEXEEV[a], T.D. SCHEPETINA[a], V.A. STUKALOV[a], S.A. SUBBOTIN[a], I.E. SHAMANIN[b], V.M. BELYAEV[b], V.I. VASYUKOV[b], K.B. VESHNYAKOV[b], V.I. ZALUGIN[b], E.B. GORYUNOV[b], YU.K. PANOV[b], V.I. POLUNICHEV[b]

[a] Russian Research Centre “Kurchatov Institute”
Moscow, Russian Federation

[b] State Unitary Enterprise "OKB Mechanical Engineering"
Nizniy Novgorod

Abstract. The future of human society and that economic situation which will be developed in the world will depend substantially on availability, quantity and quality of energy, a way of its production, and consequently also shares of nuclear energy (NE) in the general balance of manufacture of energy. Showing its attitude to NE, the society now forms not only the future of NE, but also its own future

1. INTRODUCTION

The future of human society and that economic situation which will be developed in the world will depend substantially on availability, quantity and quality of energy, a way of its production, and consequently also shares of nuclear energy (NE) in the general balance of manufacture of energy. Showing its attitude to NE, the society now forms not only the future of NE, but also its own future

NE was created on an image and similarity of organic power (OP). Its fuel cycle (FC) includes a finding of deposits of power resources in a nature; their extraction, transportation; processing (conditioning); burning (fission); removal and storage of waste products. And only regarding removal and storage of wastes NE strongly differs from OP. Fuel cycle of OP is closed by nature: gases released in atmosphere and are assimilated by wildlife; the dust, ashes, slags also are removed in an environment and in some extent are assimilated too.

FC for NE does not become isolated from natural environment though environment also may be used for storage of waste products of NE. Waste products of OP at modern scales of its use revolt global natural processes in large extent. Waste products of NE are small in volume and directly are not capable to influence in some significant degree global processes in an atmosphere, hydrosphere, lithosphere. But they may influence in some measure evolution of alive matter, especially in long-term prospect.

At usual structure of NE FC its advantages connected to huge energy of fission process is used not in the full scale (it is burnt 0,5 % of extracted uranium, at uranium contents in ore of 0,1 % for manufacture of 1 GW year of electric power is burnt 1 t of uranium, is extracted 200 t of uranium and moved 200 thousand t of ores). But the negative moments are shown in full scale: risk of non-authorised use of nuclear materials; an opportunity of catastrophic failures; radioactive pollution in places of extraction of uranium, in the NPPs sites and FC enterprises.

Modern NE, not only has taken advantage of a nuclear fuel cycle developed for defensive branches, but also has adopted practically too the attitude(relation) to recycling waste products by a residual principle. By it in many respects are caused both its(her) today's troubles and lacks.

From the very beginning of development the nuclear power became a subject of haste and competition both on the defence, and on peace directions of development. In result the basement of this new and serious technology was not properly generated: ways of creation of nuclear power installations (NPI) were adjusted, but ways of NPIs decommissioning and elimination of waste products from an environment are extremely poorly realised.

In other words, the system of nuclear power now is not a system - it is not complete and is not closed, because opportunities of its influence on natural cycles are not excluded, closed fuel cycle is not realised, transmutation of dangerous radio nuclides is not implemented.

Besides orientation to increasing up of individual power capacities of power units has resulted in their bad competitiveness in the market of investments in comparison with modern power units on organic fuel.

But it is necessary to note, that NE having lost in scales, has considerably overcome others power technologies in the way of analysis of mistakes and in the field of advances for the future both theoretical, and technical, technological, design.

If further NE would be developed only on thermal nuclear reactors its energy potential within the framework of prospective stocks of uranium will not exceed the tenth part of that may be provided due to petroleum and gas. Large scale development of new power technology is meaningful only in the case if its potential exceeds that stocks of petroleum and gas. At creation of the closed nuclear fuel cycle and at use of breeder reactors in NE system it is possible to receive amount of energy on the order exceeding potential of resources of gas and petroleum. In this case NE not only solves problems of sustainable development, but also in the long term perspective becomes attractive to long-term investments.

Changes in a world power structure show growing value of low power capacity units in XXI century, in particular for regions in which there are no advanced electric grids. Also universal tendency to decentralisation of energy supply of consumers in large power grids is traced. Large enterprises (for example, aluminium combines of Bratsk, Irkutsk, Krasnoyarsk), and the enterprises of small patterns of ownership (for example, large hospitals, hotels) prefer to refuse intermediary services of electricity grids and to use its own energy sources.

For maintenance of power safety of remote northern and Far East territories of Russia creation of a network of independent nuclear stations of small power capacity (SNPP) is especially actual. Therefore in substantive provisions of nuclear power development strategy of the Russian Federation for the period till 2010 the task of creation of SNPP system with infrastructure supporting their development and functioning is written down.

In INPRO frameworks the task of a substantiation of such system can be one of the leading if to take into account interests of developing countries – participants of this international project.

2. THE SECOND STAGE OF NE DEVELOPMENT

After two serious accidents on NPPs and loss of competitiveness in the market of investments from the beginning of 90th years there has come the second stage of NE development described by reassessment of tendencies and prospects of NE, search of new ways and ways of decisions of problem of safety and meeting of non-proliferation principles and an ecological acceptability. There is a revision of the basic approaches of creation of new generation of power reactors for future NE system:

— decrease of specific core power rate;
— decrease individual power capacity;
— monoblock designs with integrated configuration of first reactor circuit;
— increase of safety as due to reduction of danger (decrease of power capacity and quantities of radio nuclides; a care from use of toxic, combustible and explosive materials), and due to perfection of a principle defence in depth (use of various active and passive systems of control, heat removal and radio nuclides mitigation).
— designing measures allowing to the greatest degree to minimise an opportunity of heavy accidents;
— integrated approach to estimation of profitability and competitiveness;
— paying of the greater attention to problems of nuclear fuel cycle, radwaste management and NPP decommissioning;
— designing of reactors for functioning in multicomponent NE systems – thermal reactors of increased safety, fast breeder reactors and burner-reactors for transmutation of dangerous long-living radioactive waste products.

Now it is impossible to allocate any one of directions of nuclear technologies development which would solve all problems facing NE. Now there is a change of the attitude to various NPI, first of all because of influence of the safety factor, and to NE as a whole - under influence of ecological aspects and questions of non-proliferation of nuclear weapon.

The evolutionary direction of development is typical for large power capacity NPPs. The innovational way of development with wide use of new decisions and passive safety systems is characteristic for units of average and small power capacity.

Researches have shown, that the concept of inherent safety of nuclear reactors is more easy realised for units of average and small power capacity for account, in particular, opportunities of decrease of core power density. Estimations show, that decrease of unit power capacity in 10 times (with 1000 MW up to 100 MW) allows to decrease risk of heavy accidents in 1000 times. Simultaneously at the small linear sizes for these power units it is easier to realise a module design principle, to simplify control and safety systems; they have small quantity of stored heat and radio nuclides. In its power niche NPPs of small and average power capacity can compete to the traditional large power capacity NPPs on economic parameters.

Among projects of future reactors with high potential of inherent safety preference is given to small reactors cooled by liquid heavy metals (Pb-Bi), molten salt reactors (both as solid fuel and liquid fuel as burner reactor), gas cooled and light water reactor types.

2.1. The concept of holistic system of nuclear power of small power capacity

System as a whole and its separate elements should meet the requirements, developed in frameworks of INPRO for future NE system. For decision of the task of creation of complete system of NE of small power capacity the international cooperation of interested countries and large companies can be organised with the minimal investment risk for demonstration models of separate elements of the system.

In Russian Research Centre "Kurchatov Institute" the development of holistic system approach is conducted now for forthcoming large-scale development of SNPPs network.

It is supposed, that the NE structure will consist of two parts:

— ***external*** - networks of SNPPs, having the highest level of safety which operation does not demand the extreme qualification of operators, providing with energy the various technological manufactures connected to them (power sources, the built-in in technological or economic processes);
— ***internal*** - closed from "world" in which these installations are created and processed after their decommissioning and where all works on manufacturing and processing of fuel and the transmutation of dangerous radio nuclides will be carried out).

Such holistic system approach to association of different types of autonomous SNPPs on uniform industrial base of creation, service and recycling of SNPPS has as the basic purpose the increase of safety not only for SNPPs (already designed as safe), but increase of safety for back end of life cycle – decommissioning (designed as regular procedure) and especially management of spent fuel and radwaste.

Thus *the external* part of system seen by all is represented as complexes of technological enterprises on the basis of autonomous NPIs.

In such places application of SNPPs in structure of technological enterprises *complexes* – TEC – sells for manufacture of electricity, and heat, fresh water, accompanying chemical products from sea water, seafoods, hydrogen for power and technological application, gasification of coal, extraction and processing of minerals is considered. It is possible creation of comfortable places in droughty areas at coast of seas, and use of the SNPPs as a driver for gas pumping stations with a goal of gas economy and preservation of the natural environment.

As an initial basis of development *of an external* part it is now offered to consider reactors of low power. Experience of development of such reactors within more than 40 years has shown their potential opportunities by way of safety.

Design bases of NPI for SNPPs have been created for a long time. In Russia there is 40-years experience of creation and operation of transport NPIs of various types for icebreakers, submarines and space ships. There is a set of projects of multi-purpose NPIs for manufacture of heat, the electric power and cogeneration of them; various methods of construction and operation (mobile, floating, stationary) are worked out.

But remains unsolved very big complex of the problems connected to large-scale use of SNPPs in civil power.

For this purpose it is necessary to implement a complex of the basic works including study and optimization of integral structure of SNPPs system. It should include: development of system organising elements and optimization of their characteristic; creation of appropriate industrial capacities; the enterprises for operation, repair and decommissioning of SNPPs; the enterprises of a fuel cycle; the enterprises for processing of spent fuel and radwaste management (internal part of system).

It is necessary to note, that the majority of SNPP projects existing now may not answer perspective norms and requirements of the future where these projects are aimed.

With this purpose it is necessary to outline a complex of norms, criteria and requirements for perspective NPIs. Together with regulatory bodies the work on formation of regulatory bases for nuclear power system with SNPPs should be carried out.

Concerning a set of types and capacities of NPIs they should be determined by the requirements showed by society to NE system as to a basis of sustainable development. These are reactors as with thermal (for expansion of sphere of given services), and with fast (for effective breeding of fuel) spectra of neutrons, and so-called burner reactors for transmutation of dangerous radio nuclides.

Power capacity of SNPPs can be chosen anyone in an interval 1, 5, 50 and more MW (el), according to needs of definite region.

Using possessed experience and industrial base of NPIs for ice breakers and submarines, it is possible to design and adjust serial production of transportable NPIs of given power capacities. The perspective capacity of the Russian market only for these installations is estimated approximately in 2000, 1000 and 100 units accordingly (about 20 GW).

Low core power density lays in a basis of the concept of autonomous power reactor, allowing to ensure the functioning without refuelling during 8 - 15 years and more.

Reactors of low power should be transportable at practically full factory readiness. For such reactors integrated modular configuration is most expedient. They entirely can be made and tested industrially and in gathering are delivered to an operation site. It essentially reduces terms and expenses for construction and creation of an additional infrastructure on the operation sites, raises quality of manufacturing and reliability of NPI, facilitates a decommissioning problem.

At such approach the user will not have problems with short-term changes of cost of the electric power in the world market, there will be no cares with decommissioning and radwaste management. Upon termination of operation, NPI will be replaced by new one (similarly to electric sells – nuclear batteries), and spent NPI will be delivered for processing in *the "internal"*, protected part of NE structure.

The Internal part of NE system also should be organised in special manner - several large but compact enterprises on which all radioactive dangerous technologies and operations with NPI and fuel are concentrated. These special strictly protected enterprises can be named central repair - reloading bases (CRRB).

Functions of this internal part of SNPPs system should be provided by specially created (or reconstructed existing) enterprises - regional CRRB, taking place, for example, in territory of Russia and providing a full spectrum of service for SNPPs, located in the different countries of region.

As a technical basis *of "internal"* part it is offered to use non aqueous, for example, fluorine gas volatility technologies for spent fuel processing and molten salt burner reactors, able to work with any nuclides compositions even in sub critical mode with external neutron source.

These technologies allow to use effectively all actinides, what does not leave them unrequited by NE system and makes them poorly accessible to terrorists, thus facilitating the decision of a problem of non-proliferation of nuclear materials.

3. PROJECTS OF SMALL NUCLEAR POWER INSTALLATIONS

There is many projects of small Nuclear Power Installations (NPI) (up to 50 MWe.). Among them are most known KLT-40, ABV (pressurised water); RUTA, ELENA (water with natural circulation); ANGSTREM, TES-M (coolant lead-bismuth).

In OKB Mechanical engineering a long-term experience of designing NPI with pressurised water reactors of various type for ships and submarines is saved up, positive results of their operation and technical perfection are received.

For immediate prospects on the basis of available in OKBM scientific and technical work already done are recommended to realisation NPI with elements fulfilled in test facility conditions and in conditions of real operation by that allows at the minimal expenses and in deadlines to create installation of new generation with qualitative more high safety level and technical and economic parameters.

To the remote prospect in OKBM the conceptual projects, based on the original design decisions are studied, allowing to provide highest technical and economic parameters SNPP on the basis of ship technologies and its appeal to the customer as power source of concerning small capacity.

For example, for the perspective nuclear ice breaker the following types of power installations are considered:

- Loop type units (Figure 1), completely focused on use of the modern equipment, went bench and technological tests and operation check at working objects;
- The integrated type units, using the modern equipment or its base elements which has been tested.

Nuclear power installation ABV-6M having the final design, NPI KLT-40S with the developed and coordinated design project and NPI on the basis of serial installation KN-3 may be recommended for realisation now as a safe nuclear power sources for NPP of small power with acceptable technical and economic parameters.

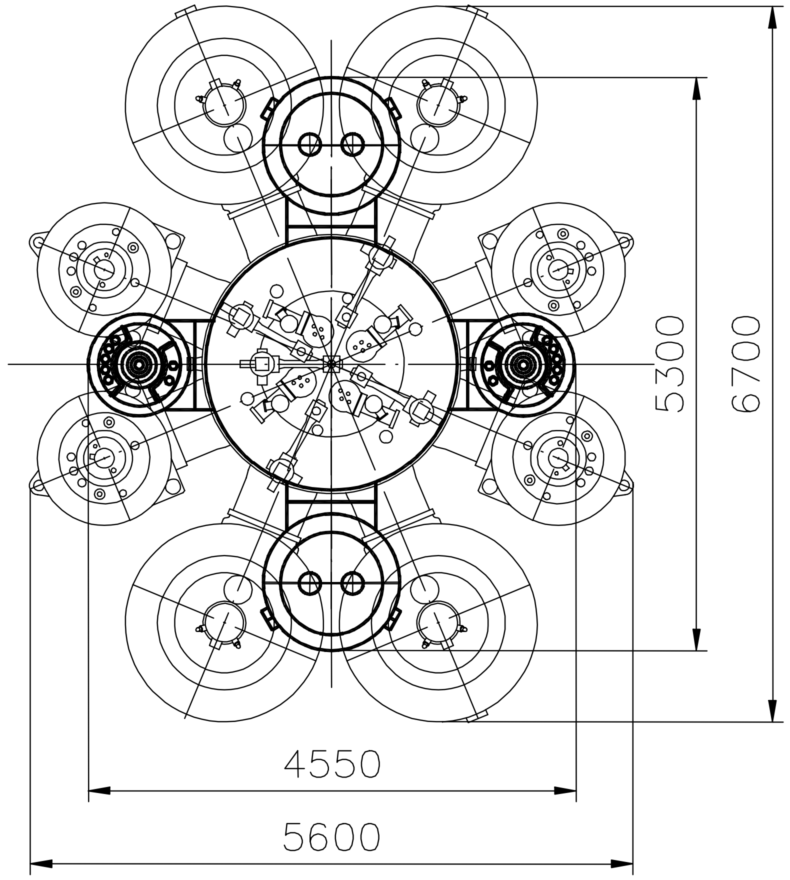


FIG. 1. Size comparison of NPI OK-900A type and of new generation 200 MW capacity.

3.1. Cogenerating nuclear power plant with KLT-40S reactors

Now in Russia works on the project floating Cogenerating NPP (CNPP) (Figure 2) on base of floating power unit (FPU) with installations KLT-40S are most advanced. For today design projects of NPI and FPU are developed and protected, the license of GAN of the Russian Federation for equipment manufacturing for CNPP is received. CNPP Project has successfully passed the State ecological examination. Licenses for accommodation and construction CNPP are received.

FPU - basic component of CNPP, it represents smooth-deck non-selfpropelled ship with the inhabited and power module. The structure of the power module includes two NPI KLT-40S, two steam turbine installations and electropower system. At the FPU storehouses of spent fuel assemblies, liquid and firm radioactive waste products are placed. FPU has an own complex for reactors reloading. All operations with radioactive substances will be carried out only at FPU.

The floating power unit maintenance at the specialised ship-building enterprise and is towed to a place of operation in completely ready kind. For care regenerative and repair work delivery of FPU to the specialised enterprise is stipulated.

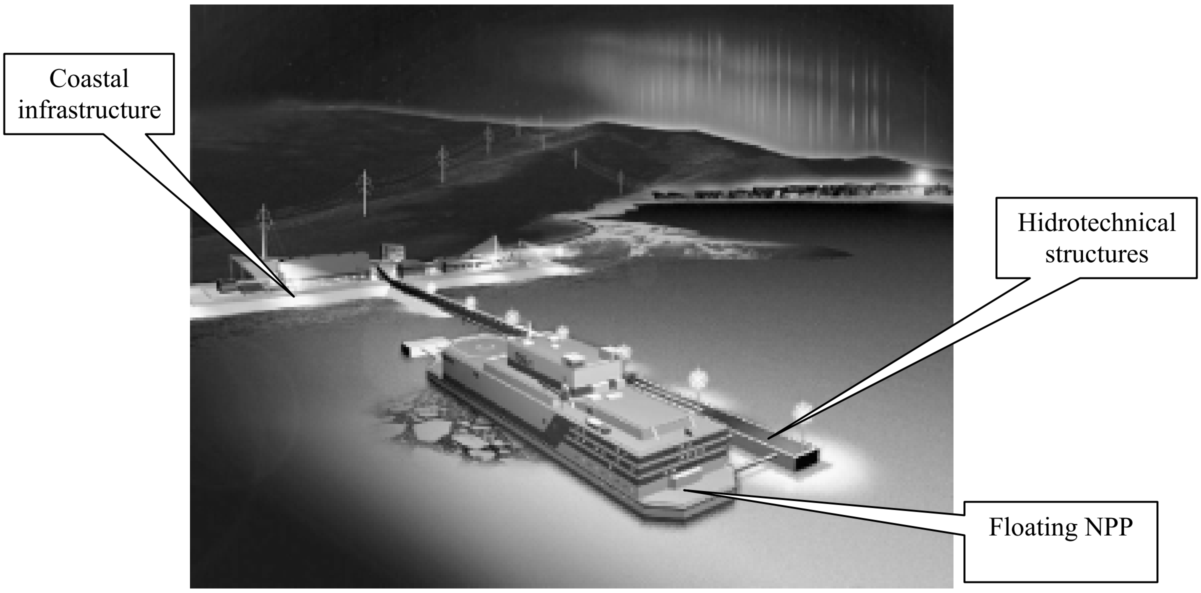


FIG. 2. Floating CNPP.

At decommissioning FPU is transported to its recycling and a burial place with preservation of "the green lawn" condition of CNPP operation area.

3.2. Nuclear power installation KLT-40S

NPI KLT-40S is created on the basis of serial ship NPI as KLT-40 type, having an accident-free operating time more than 200 reactor-years. Components of installation are modernised with the purpose of increase of reliability, resource, service life and improvement of maintenance service conditions. Safety stems are lead up to modern requirements of GAN of Russia normative documents for nuclear plants at basing on ship.

The basic nominal characteristics of KLT-40:

— Thermal capacity - 150 MW;
— steam production - 240 t/h;
— Pressure of the first coolant - 12,7 MPa;
— Steam pressure - 3,72 MPa;
— Temperature superheated steam - 290°C;

Parameters of reliability:

— Service life - (35-40) years
— Between-repairs period - (10-12) years.
— Resource of the irreplaceable equipment - (240-300) thousand h.
— Resource of the replaceable equipment - (80-100) thousand h;
— The period of continuous operation - 8000 h.

Reactor KLT-40S being advanced analogue of installations of nuclear ice breakers, may be created at the minimal expenses and in the deadlines.

3.3. Nuclear cogenerating power plants on the basis of other installations [3]

The high level of study (final NPI design is executed and coordinated with Gosatomnadzor of the Russian Federation) has also CNPP on the basis of two NPI ABV-6M, developed at variants of ground (stationary or modular) and floating design.

Distinctive feature of ABV-6M NPI (Figure 3) is containing of reactor core and steam generators in the vessel of an integrated pressurised water reactor with the born pressurising gas system, an all-modes coolant natural circulation. By development NPI ABV-6M the experience of creation as transport and stationary power installations is used.

For the variant of ground modular CNPP design the opportunity of delivery both reactor and turbine-generator modules (diameter of 8,5 m, length of 12 m and weight 600 t) is stipulated by marine-automobile transport (Table I).

At the stage of the outline design it is developed power line reactors such as CNPP (Figure 4) from 50 up to 230 MW (e).

At a basis of the project the pressurised water reactor of integrated type contained in the safety vessel with internal steam-gas pressurising system and all-mode natural circulation of the coolant is fixed.

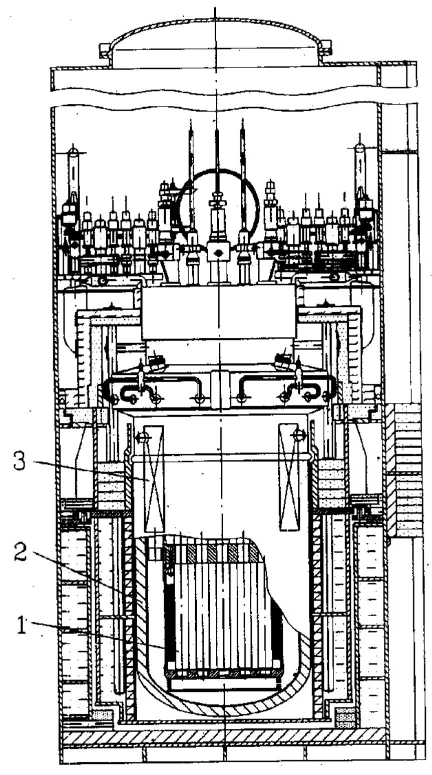
As example of independent (without permanent service) power source with long (till 25 years) service life without reactor reloading is an outline project NPI SAHA-92 developed on the basis of a pressurised water reactor of integrated type (Figure 5) with internal steam-gas pressurise system.

Use of conversion development elements (a tight asynchronous turbine-generator, a tight second circuit, including pumps) allows to refuse number of providing systems and to decrease number of the personal (down to care of periodic service).

Table I. Technical and economic parameters of CNPP

Parameter	**Value**			
NPI type	**SAHA-92**	**ABV-6M**	**KLT-40S**	**ATEC-80**
Electric capacity, MW	1	2x12	2x35	2x85
Heating thermal capacity, Gkal/h	1-2,6[1]	24[1]	50[1]	112
Capital expenses for construction, million $	20	110	165	430
The boundary tariff on el./energy, cent /kWh	26,1	7,2	4,1	3,5
The boundary tariff on term./energy, $/GKal		32,8	8,13	6,3
Service life, years	25	50	40	60

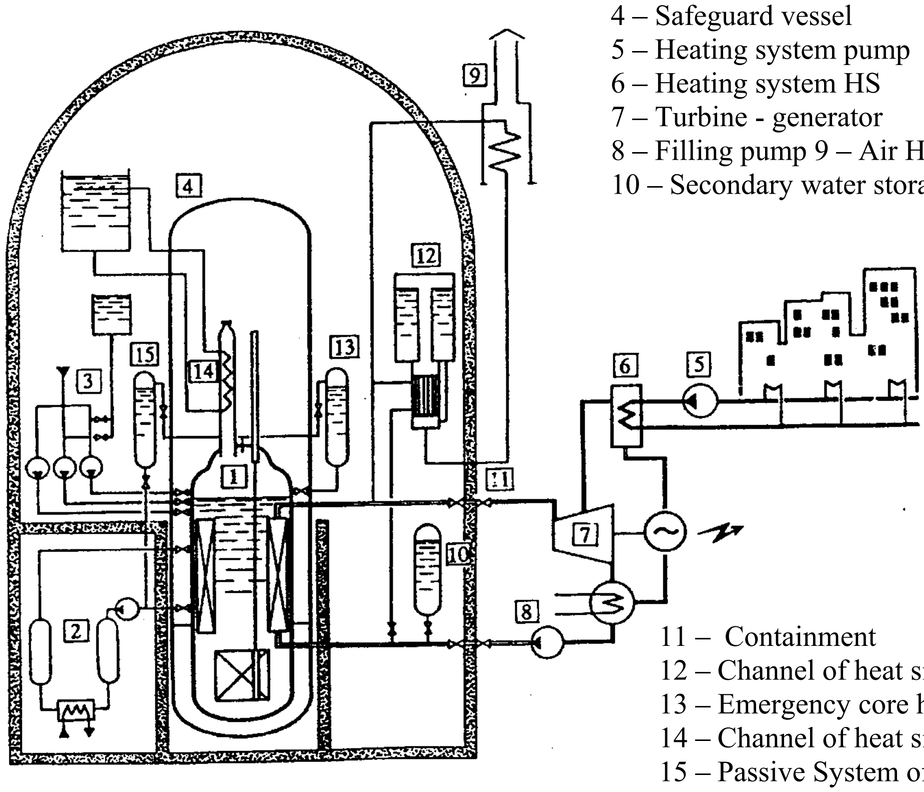
[1] At reduction of electric capacity



1 – Core
2 – Reactor vessel
3 – Steam generator

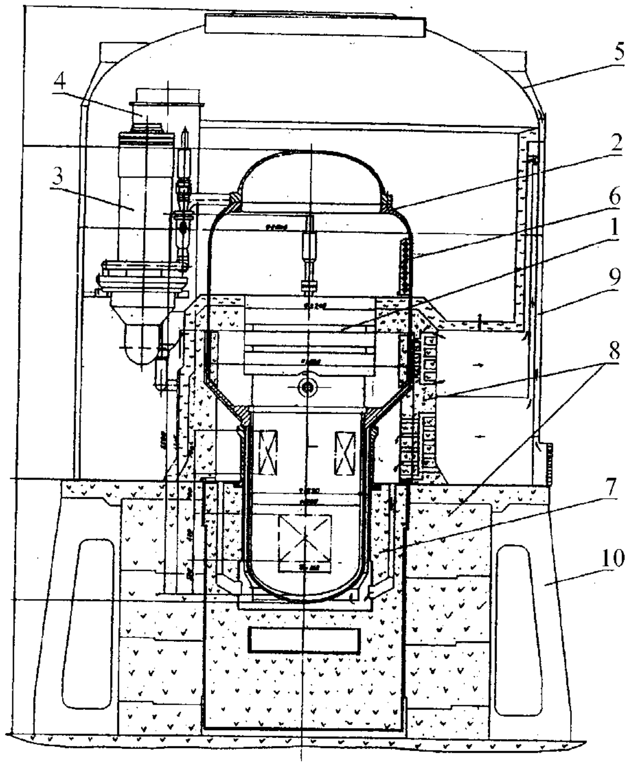
FIG. 3. Reactor installation ABV-6M.

1 – Reactor 2 – Cleaning system
3 – System of refilling water and boron
4 – Safeguard vessel
5 – Heating system pump
6 – Heating system HS
7 – Turbine - generator
8 – Filling pump 9 – Air HX
10 – Secondary water storage



11 – Containment
12 – Channel of heat sink 2-circuit
13 – Emergency core heat sink system
14 – Channel of heat sink in reactor
15 – Passive System of boron inject

FIG. 4. Cogenerating NPP.



1 – Reactor
2 – Safeguard vessel
3 – Turbine - generator
4 – Condenser
5 – Containment
6 – HX
7, 8 – Bio-protection
9 – Air cooling
10 – Support structure

FIG. 5. Nuclear power installation SAHA-92.

3.4. Ways of economic efficiency increase

Actual technical and economic problem is reduction of project recovery times, increase of the income and net profit for the CNPP operation period which in short enough terms may be successfully decided due to increase of individual capacity of NPI KLT-40S, or due to use of the most powerful in Russia ship propulsion reactor such as KN-3, having high compactness of the equipment accommodation and created for the Navy ships on the basis of experience of development and creation of ice-breakers fleet nuclear installations.

At OKBM the analysis of an opportunity of NPI KLT-40S and KN-3 capacity increase was carried out on the basis of introduction of the fulfilled decisions under condition of installation shape preservation.

The key decision is use of cassette core of low fuel enrichment in reactor that answers conditions of non-proliferation nuclear materials and opens ample opportunities for exit power sources of small power with specified NPI on the foreign market.

At the minimal modernisation of installations the nominal CNPP electric capacity on the basis of two NPI KLT-40S and KN-3 is estimated by size 120 MW and 210 MW, and heating - 84 GKal/h and 150 GKal/h, accordingly. Thus the cost price of the electricity of modernised NPI KLT-40S is estimated by size about 3 cent / kWh.

Thus, the modernised NPI and CNPP as a whole having the much greater capacity in comparison with base ones, have higher technical and economic appeal to regions with the appropriate level of energy requirement.

3.5. *The conceptual project of NPI with an integrated type reactor*

As a perspective NPI at the greatest degree meeting the requirements, showed to power units of small power, is developed the conceptual project of the power unit by electric capacity 60 MW on the basis of a pressurised water reactor of integrated type of thermal capacity 200 MW (Figure 6).

The core with the increased life time, lowered power density and low enrichment on U235 is applied. The sizes of a steam generator, the core, internal steam-gas pressurise are optimised with the purpose of reception of best NPI characteristics.

Presence of a significant stock of the coolant above the core, the powerful channel of heat sink through steam generator, natural circulation of primary coolant, connection of external pipelines of primary circuit to reactor vessel in its top part through throttle devices, installation of external systems cutoff armatures directly on the reactor vessel allow essentially increase NPI safety and self-protection in emergency operation.

Safety systems functioning principles are accepted in view of floating CNPP arrangement.

Capital costs for creation double-reactors floating CNPP and the cost price of the electric power are estimated by sizes 188 million $ and 2 - 2,2 cent / Kwh accordingly.

At preservation of a reactor transportability by rail, the variant with the increased sizes of the core is developed, allowed to increase thermal and electric capacities up to 330 and 100 MW accordingly, with improvement of technical and economic parameters.

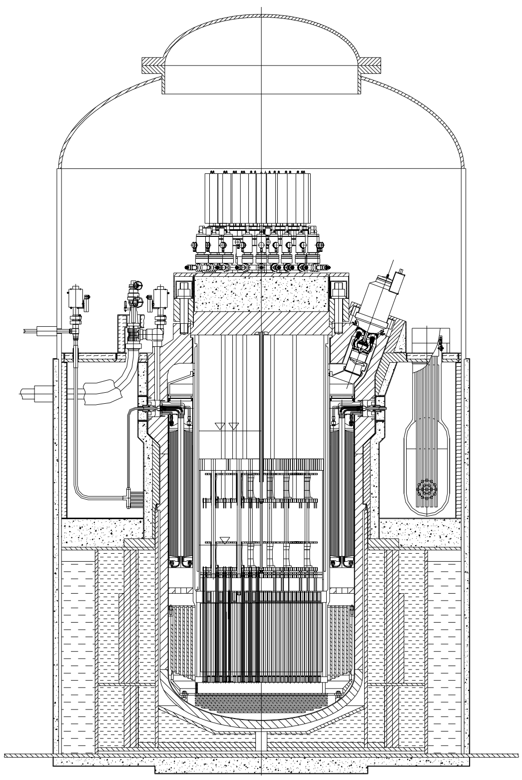


FIG. 6. Reactor RIT with internal steam-gas pressurise system.

3.6. NPP for sea water desalination

Now under orders of Ministry of Atomic Energy of Russian Federation OKBM develops the project floating nuclear power water desalination complex (FNDC) with the purpose of interface optimisation of various types NPI of pool and integrated design and various types desalination installations.

Basic components of FNDC are FPU which structure similar CNPP, and floating desalination block (FDB) which structure includes desalination installation. Thermal energy and a part of electric energy produced by FPU is used for realisation of desalination process, other electric power is released to consumers.

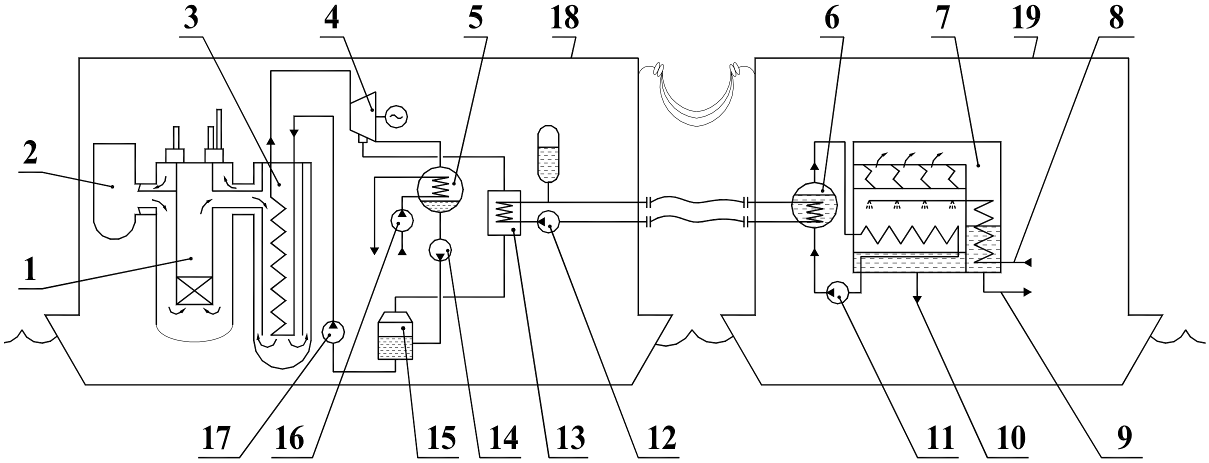
At FNDC project the most advanced and economic desalination technologies with use reverse osmosis and distillation desalination installations are used.

Basic FNDC schemes with distillation and reverse osmosis desalination installations are given in Figures 7 and 8.

Basic comparative technical and economic characteristics FNDC on base FPU with various NPI types and floating desalination block with distillation (DI) and reverse osmosis (RO) desalination installation, are submitted in the Table II.

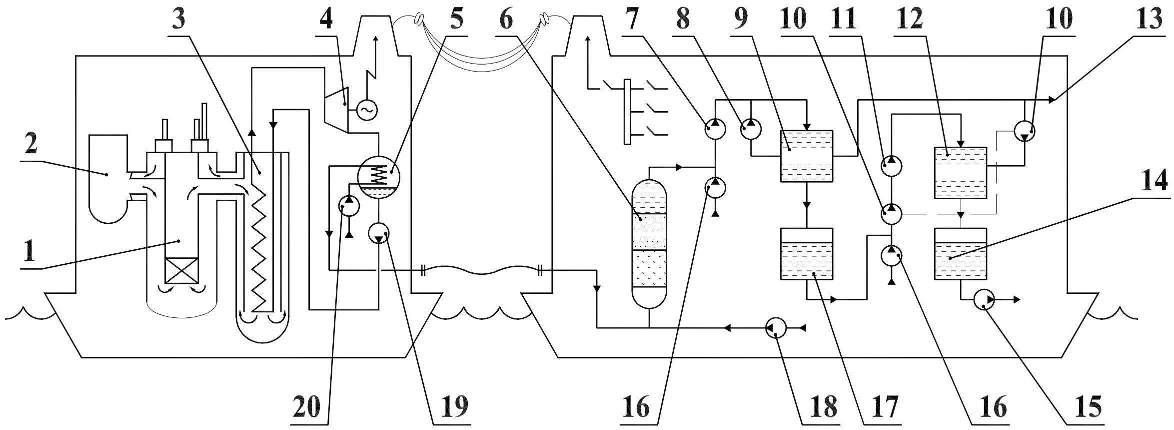
Table II. Technical and economic parameters of CNPP

The name of parameter	Type NPI							
	KLT-40S				KN-3		RIT	
Quantity of NPI	2		1		1		1	
Thermal capacity, MW	2x150		150		300		150	
Desalination installation type	DI	RO	DI	RO	DI	RO	DI	RO
Electric capacity FPU, MW	70	77	35	38,5	75	75	35	38,5
Fresh water capacity, m^3/day	2x40000		40000		2x40000		40000	
Electric capacity at DI work, MW	2x21	2x29	21	29	36,1	52,5	21	29
Capital costs for FNDC construction, million $	331	295	181	162	293	257	175	156
The boundary tariff for the electric power, cent / kWh·	4,71	4,26	5,09	4,60	3,90	3,53	4,66	4,22
The boundary tariff for the freshened water, $/$m^3$	0,88	0,70	0,94	0,75	0,90	0,67	0,90	0,73



1 – reactor
2 – main pump
3 – steam generator
4 – turbine –generator
5 – condenser
6 – SG of desalination installation
7 – distillation of desalination installation
8 – marine water
9 – fresh water
10 – brine
11 – pump
12 – intermediate circuit pump
13 – intermediate circuit heater
14 – condensate pump
15 – deaerator
16 – pump
17 – pump
18 – FNPP unit with KLT-40C
19 – floating desalination unit

FIG. 7. Common scheme of desalination complex with distillatory installation.



1 – reactor
2 – 1 circuit pump
3 – steam generator
4 – turbine - generator
5 – condenser
6 – preliminary filter
7 – middle pressure pump
8 – recirculation pump
9 – membranes super-filtration
10 – energy regenerating system
11 – high pressure pump
12 – reverse osmosis membranes
13 – exit of brine
14 – filling water vessel
15 – fresh water exit pump
16 – entrance of reagents
17 – super-filtration vessel
18 – main pump
19 – condensate pump
20 – marine water pump

FIG. 8. Common scheme of desalination complex with reverse osmosis installation.

4. CONCLUSION

Thus, the experience of small power capacity NPI development not only allows to meet user requirements in autonomous reliable and safe energy sources on quite competitive basis in conditions of the big commercial interest to them, but also allows NE to go through the period of stagnation of interest to it in large energy systems without loss of potential in the nuclear technologies especially concerning creation of a fuel cycle for future large-scale NE system.

SNPPs system can become experimental site for improvement of technical decisions which may be used for sustainable development large scale NE system.

As it is new system, and it only begins to arise, it is meaningful to use in its structure only the best achievements of scientific and technical progress as regarding increase of efficiency of production, transformation and use of energy, and for decision of safety issues, non-proliferation, an ecological acceptability and economic attractiveness.

Creation of complete SNPP system will demand essential investments of material means and significant intellectual efforts, excessive within the framework of one state.

For regions of energy deficits it is offered to develop SNPP system in a complex with CRRB on the basis of interstate cooperation.

It is possible to draw a conclusion, that in modern conditions study of a wide complex of the problems connected with SNPPs, and allocation of this direction of NE in important section of power researches for the international cooperation is actual enough.

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FUEL CYCLE OF BREST REACTORS. SOLUTION OF THE RADWASTE AND NONPROLIFERATION PROBLEMS

A.G. GLAZOV[a], A.V. LOPATKIN[a], V.V. ORLOV[a], V.I. VOLK[b], P.P. POLUEKTOV[b], V. F. LEONTYEV[c], R. S. KARIMOV[d]

[a] Research and Development Institute of Power Engineering (NIKIET)
Moscow, Russian Federation

[b] Research Institute of Inorganic Materials (VNIINM)
Moscow, Russian Federation

[c] Special State Design Institute
Moscow, Russian Federation

[d] SverdNIIkhimmash
Ekaterinburg, Russian Federation

Abstract. Fast reactors with a nitride fuel and a lead coolant (BREST) have low excessive in-core plutonium breeding (CBR ~1.05) and do not have breeding blankets. The fuel cycle of BREST reactors includes stages that are traditionally considered in a closed fuel cycle of fast reactors excluding the breeding blanket cycle, namely in-pile fuel irradiation, post-irradiation cooling of spent FAs (SFAs); SFA transportation to the recovery shop, SFA dismantling, fuel extraction and separation of the SFA steel components, radiochemical treatment, adjustment of the fuel mixture composition, manufacturing of nitride pellets, manufacturing of fuel elements and fuel assemblies, interim storage and transportation to the reactor. There is a radioactive waste storage facility at the NPP site. The fuel cycle of fast reactors with CBR of ~1 does not requires plutonium separation to produce "fresh" fuel, so it should use a radiochemical technology that would not separate plutonium from the fuel in the recovery process. Besides, rough recovered fuel cleaning of fission products is permitted (the FP residue in the "fresh" fuel is 10^{-2}-10^{-3} of their content in the irradiated fuel) and the presence of minor actinides therein causes high activity of the fuel (radiation barrier for fuel thefts). The fuel cycle under consideration "burns" uranium-238 added to the fuel during reprocessing. And plutonium is a fuel component and circulates in a closed cycle as part of the high-level material. The radiation balance between natural uranium consumed by the nuclear power closed system and long-lived high-level radioactive waste generated in the BREST-type nuclear reactor system is provided by actinides transmutation in the fuel (U, Pu, Am, Np) and long-lived products (Tc, I) in the BREST reactor blanket and by monitored pre-disposal cooling of high-level waste for approximately 200 years. The design of the building and the entire set of the fuel cycle equipment has been completed for a BREST-OD-300 experimental demonstration reactor, which will implement the basic features of the BREST reactor fuel cycle.

1. INTRODUCTION

Large-scale nuclear power – the industry that can meet about half of the future demand for electricity production – must comply with a set of criteria for safety, economics and fuel cycle strategy. This paper discusses the BREST fast reactor fuel cycle concept from the viewpoint of its specifics and ability to meet the requirements for nonproliferation of nuclear materials and to establish a balance between the generated radioactive waste and the mined natural uranium.

Design premises for the fuel cycle of BREST reactors:

- Periodic fuel reprocessing and fabrication in a closed cycle.
- Full Pu reproduction in the core without U blankets and with BR~1.05.
- Transmutation of most hazardous long-lived actinides (as part of fuel) and fission products (irradiation outside of the core). Profound cleaning of radioactive waste to remove these nuclides. Radiation balance between buried RW and uranium mined from earth.
- Rough fuel cleaning from fission products during reprocessing. Fuel facilities in the closed cycle should be unsuitable for Pu recovery from spent fuel (technological support to nonproliferation).
- On-site fuel facilities to avoid shipment of large amounts of high-level and fissile materials.
- Cost-effectiveness of the entire complex (reactor and fuel cycle).

The BREST-OD-300 NPP design includes the plant proper with a demonstration liquid-metal reactor BREST 300 MWe in capacity, the on-site closed fuel cycle and the complex for radwaste treatment and storage. The design studies have confirmed the feasibility of building BREST reactors of various capacity (e.g. 600 and 1200 MWe) for the large-scale power industry of the future, following the same principles as those designed into the 300 MW reactor. The BREST-OD-300 facility is a pilot, demonstration power unit meant to validate and further develop the design features adopted both for the reactor facility and for the on-site fuel cycle with a radwaste management system. On completion of the essential studies, the power unit is to go into commercial operation in the grid. Subsequent commercialisation is expected to proceed with the NPP comprised of two BREST-1200 units and having an on-site fuel cycle, which has gone as far in its development as a full-fledged conceptual design (Figure 1).

According to current expectations, the BREST-1200 plant design will rely on the BREST-OD-300 developments tried out in operation: fuel rods and assemblies, basic equipment of the plant proper and its on-site cycle. Transfer to the higher installed capacity will be achieved largely by increasing the number of the tried-out components. Thus, the transition from the pilot plant to a commercial facility may be effected with minimised time and money spent on it. It should be also mentioned that the on-site fuel cycle will be shared by the two power units.

The BREST fuel cycle promises virtually unlimited expansion of the fuel resources available to the nuclear power industry due to recycling of U-Pu-MA fuel of equilibrium composition (CBR $\approx$ 1,05) which will require addition of but small quantities of depleted or natural uranium to compensate separated fission products [1]. The fuel cycle arrangement allows attaining the radiation equivalence of nuclear materials with allowance made for their migration. To this end, the radioactivity and the nuclide composition of the waste subject to burial should be such that the heat and the stability of the buried materials and the degree of migration risk of the nuclides, with regard to their respective biological hazards, should be at least no worse than those found at natural uranium deposits [2].

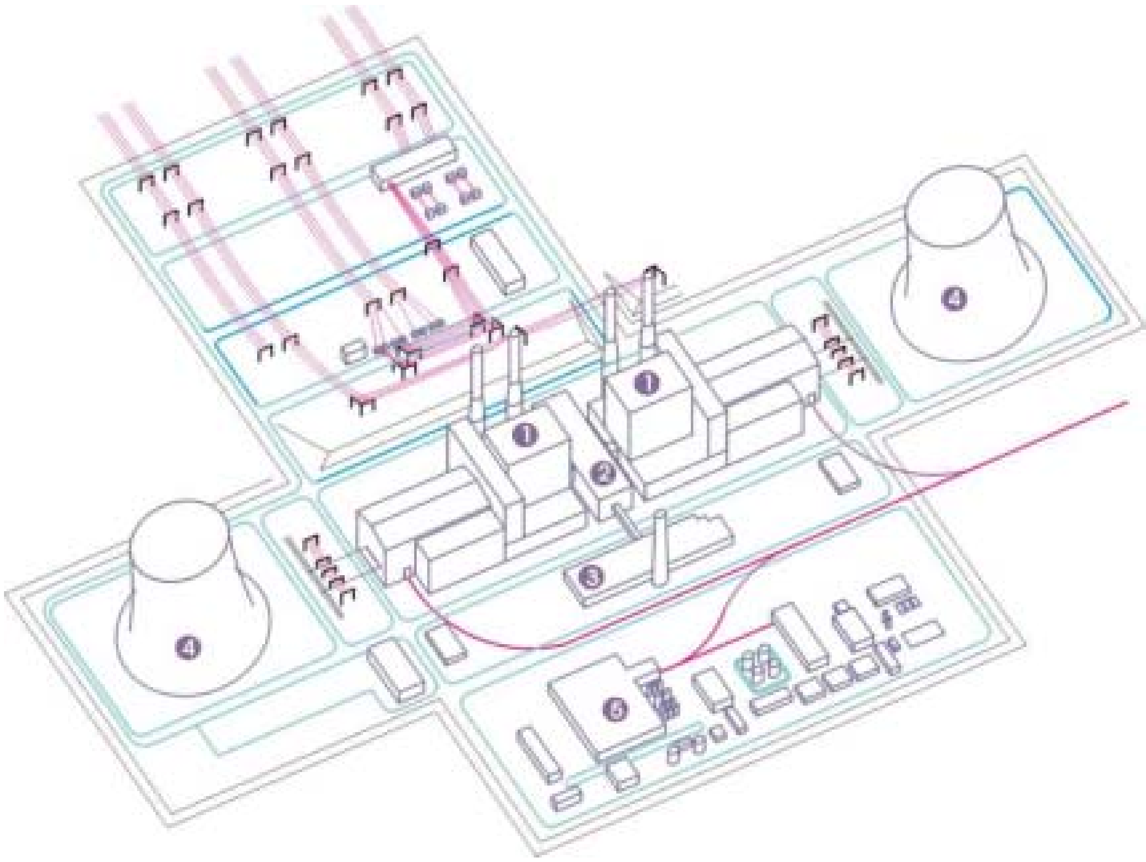
An on-site nuclear fuel cycle (SNFC) has been developed for the BREST-OD-300 reactor. Designed SNFC must provide reprocessing and fabrication of fuel of BREST-OD-300 and BN-800 reactors with nitride core (Table I).

Design documentation consists of:

— master plan, fuel cycle buildings, RW storage facilities and transportation links;
— design of equipment to be used at all stages of the fuel cycle;
— automation, communication lines and alarms;
— environmental protection;
— cost estimates.

The BREST-OD-300 fuel cycle consists of the stages usually included in the closed fuel cycle of fast reactors, except for the fuel cycle of breeding blankets:

— in-pile fuel irradiation (4-5 years);
— post-irradiation cooling (1 year) of spent fuel assemblies (SFA);
— SFA transportation to the SNFC building;
— SFA cutting to extract fuel and separate steel components;
— radiochemical treatment of fuel (reprocessing);
— adjustment of fuel composition;
— fabrication of nitride pellets;
— fabrication of fuel rods and fuel assemblies;
— temporary storage of fuel assemblies;
— FA transportation to reactor.



1 - BREST-1200 reactor building; 2 - building of the on-site closed fuel cycle;
3 - radwaste treatment and storage building; 4 - cooling tower; 5 - auxiliary buildings.

FIG. 1. General layout of the BREST-1200 NPP.

Table I. Main characteristics of the on-site nuclear fuel cycle

Characteristic	Value
Annual plan for FA reprocessing and fabrication, FA/year:	
BREST-OD-300	29
BN-800	259
Pu going to waste, %	0.5
Annual consumption of some materials and agents:	
Depleted uranium, kg/year	998
Hydrogen gas, nm^3/year	3808
Liquid nitrogen, kg/year	5000
Liquid argon, t/year	900
Helium gas, nm^3/year	100
Steel for BREST, kg/year	3000
Steel for BN, kg/year	12000
Zinc, kg/year	2430
Chlorine, potassium, lithium chlorides, kg/year	630
Staff	240
Operating power of process equipment, total kW	2240
Total area of the building, m^2	31500
Space of the building, m^3	236400
Stack releases after filtering (including RW treatment facilities)	
Individual isotopes, Bq/year:	
^{3}H	$4.09 \cdot 10^{9}$
^{85}Kr	$7.67 \cdot 10^{15}$
^{129}I	$1.22 \cdot 10^{7}$
Aerosols, total (including short-lived), Bq/year:	
α-active	$1.77 \cdot 10^{7}$
β-active	$1.49 \cdot 10^{10}$

The cycle includes collection of radioactive waste, their partition and preparation for storage. The entire process of fuel recycling takes place in the reactor building and in the adjacent building of the on-site nuclear fuel cycle. There is storage facility at the site to accommodate radioactive waste.

SNFC of BREST-OD-300 is designed for a capacity of 17.6 t (U,Pu)N/year under conditions of the first core fabrication and for ~ 3.5 t (U,Pu)N/year under conditions when the fuel is regenerated and refabricated. Main characteristics of the on-site nuclear fuel cycle are presented in Table I. Key requirements for reprocessing technique for BREST fuel is presented in Table II. The fuel cycle building layout is shown in Figure 2.

The BREST-OD-300 fuel cycle design involves electrochemical reprocessing of irradiated nitride fuel with separation of uranium+plutonium and MA in molten LiCl-KCl. The basic features of such a technology were developed at the Argonne National Laboratory (USA) and were elaborated in a whole number of efforts, including those undertaken by NRIIM. The fuel cycle equipment was designed by a special organisation of SverdNIIkhimmash.

SNFC technology involves the following main processes:

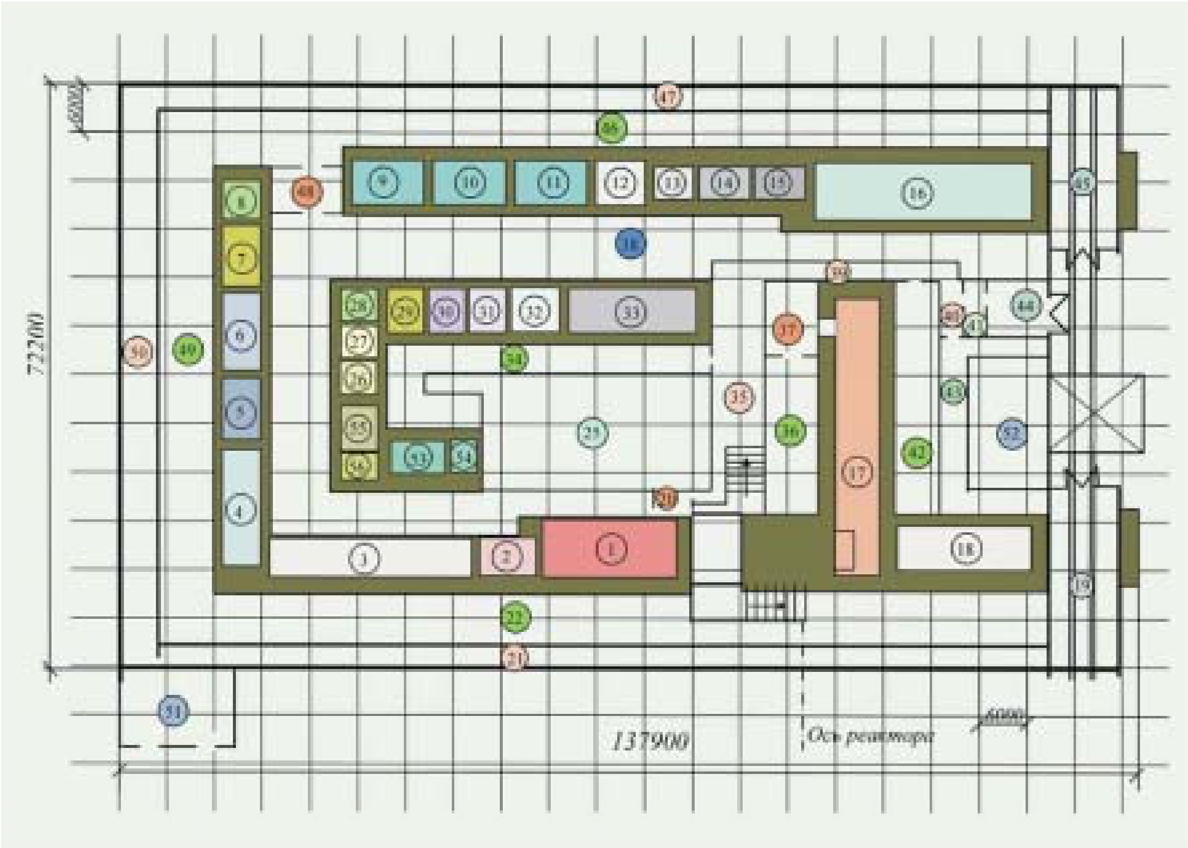
— separation of cladding and lead bond from fuel by dissolving the active part of fuel assembly in molten metallic zinc;
— preparation of LiCl-KCl salts with minimum oxygen content and melt saturation with uranium and plutonium trichlorides;
— anodic dissolution of irradiated nitride fuel in molten salts;
— sedimentation at the solid cathode of metallic U, Pu, Np, Am and some Cm;
— periodic additional separation of U and Pu from molten salts under altered conditions of electrochemical process during electrolyte recovery;
— vacuum melting of mixed metallic U-Pu-Np-Am-Cm + 10 % RE at 1000 ^{o}C;
— granulation, hydrogenation and nitration of the metallic mixture, production of nitride powder, and distillation of electrolyte to be returned to the electrolyser.

All the processes are provided with systems for cleaning the released gases from aerosols and volatile radioactive elements (tritium, iodine, krypton).

Table II. Key requirements for reprocessing technique for BREST fuel

Source material	Nitride U-Pu-MA fuel, ~9% FP
Uranium and plutonium are kept together	
RW cleaning from actinides	<0.1 % remaining in fuel
Fuel cleaning from FP	0.1-1 % remaining in fuel
End product	metal or nitride (U+Pu+Am+Np)
Sr, Cs separation from RW	1-5 % remaining in fuel
I, Tc separation from RW	1-5 % remaining in fuel
Actinide content in separated Sr, Cs, I, Tc fractions, no more than	0.1 % (at)
Cm separated from fuel	to be stored out of pile during 50 to 70 years, with Pu (Cm decay product) returned in reactor
Content of Cm in “fresh” fuel	1-10% remaining in fuel

A distinguishing feature of the fuel cycle arrangement is the unattended mode of its processes, i.e. complete remote control of the basic process, equipment adjustment, repair and maintenance.



No. in plan	Work area	No. in plan	Work area
1	Area for cutting of fuel assemblies and opening of fuel rods	21, 35, 39, 40, 47, 50	Corridor
2	Chamber for fuel preparation for regeneration	22, 34, 36, 42, 46, 49	Control room
3	BREST fuel regeneration chamber	23, 24, 38	Maintenance corridor
4	Nitride preparation chamber	25	Ventilation chamber
5	Storage of powder containers	26	Crushing chamber
6	Chamber for moulding powder preparation	27	Carbothermy chamber
7, 29	Moulding chamber	30	Blending chamber
8, 28	Reloading lock	31	Plutonium oxide storage chamber
9, 10, 11	Drying and sintering chambers	32	Laboratory
12	Storage of sintered pellets	33	Test laboratory
13	Complex for inspection of pellets and components	41	Uranium unloading compartment
14	Complex for assembly and sealing of fuel rods	43	Area for preparation of claddings and fuel rod components
15	Complex for outgoing inspection of fuel rods	44	Anteroom for components
16	Storage of fuel rods	45	Anteroom for fuel rods
17	Fuel assembly fabrication area	51, 52	Personnel access point
18	Fuel assembly handling chamber	53	Facility for decontamination of large components
19	Anteroom for BN-800 fuel assemblies	54, 56	Depot
20, 37, 48	Additional sanitary area	55	Facility for decontamination of small components

FIG. 2. Layout of the BREST-OD-300 fuel cycle facilities (plan at elevation 0.0).

2. TECHNOLOGICAL SUPPORT TO NONPROLIFERATION OF WEAPONS-GRADE MATERIALS IN THE CLOSED FUEL CYCLE OF BREST FAST REACTORS.

Fast reactors do not need enriched uranium, i.e. enrichment services may be curtailed and then given up with time. Pu and spent fuel will be gradually removed from existing storage facilities and spent fuel cooling pools at NPPs to be used for fabrication of the first cores for fast reactors (spent fuel reprocessed to recover Pu). Initial recovery of Pu and fabrication of the first cores for fast reactors should be carried out at safeguarded facilities in nuclear countries (Fig. 3).

The BREST-OD-300 design is notable for its focus on engineering rather than organisational provisions for proliferation resistance. In the fuel cycle of BREST reactors with CBR~1.05:

— all FAs of the core contain the same amount of Pu;
— there are no uranium blankets breeding weapons-grade Pu because they are not needed;
— both before and after regeneration, the BREST reactor fuel is unfit for production of nuclear weapon;
— there is no need to recover Pu for fabrication of fresh fuel (it is suffice to separate fission products and add depleted U). Hence, reprocessing may be used because it is not suitable for Pu recovery;
— there is no need for U enrichment;
— surplus Pu is used as part of U-Pu mixture for fabrication of the first core of new reactors;
— reprocessed fuel is partially cleaned from fission products (fresh fuel contains 10^{-2} – 10^{-3} FPs present in spent fuel) and incorporates minor actinides (MA), which makes fuel highly radioactive (radiation barrier to fuel thefts).

In the fuel cycle under discussion, reactors burn ^{238}U added in fuel during reprocessing. Pu is part of fuel and recycles in the closed cycle as part of highly active mixture (combustion catalyst for ^{238}U).

The fuel cycle of BREST-OD-300 reactors is arranged without transporting irradiated fuel to an external reprocessing facility. After one-year cooling in the in-pile storage, the irradiated fuel assemblies are passed on to the fuel cycle facility via a transport passageway connecting it with the reactor compartment. Thus, the design eliminates all the risks and costs related to fuel shipment for regeneration and obviates the need for the associated handling and transportation equipment.

The electrochemical reprocessing will be modified or changed in BREST-1200 SNFC design to provide the protection from plutonium extraction.

3. BREST-OD-300 RADWASTE MANAGEMENT

3.1. Attaining radiation equivalence

Pu and Am are principal contributors to potential long-term biological hazard of spent fuel (cooling time 10^2 – 10^5 years). Radiation balance between natural uranium used in the closed nuclear power system and long-lived radioactive waste produced in the system of large scale nuclear power based on reactors of the BREST type can be attained through (Figure 3):

— profound cleaning of radioactive waste from actinides;

— actinide transmutation as part of fuel (U, Pu, Am, Np, Cm) and long-lived products (Tc, I) in the out-of-core zones of BREST reactors;
— monitored storage of high-level waste during ~200 years prior to disposal to reduce their activity 1000 fold.

Management of liquid and solid radioactive wastes of the BREST-OD-300 plant is arranged in compliance with the requirements of radiation-equivalent disposal.

The generated waste may be conventionally divided into two major categories:

— low- and medium-level liquid and solid wastes which are largely typical of NPPs. They emerge in a broad spectrum and relatively large volumes;
— high-level solid waste produced in the fuel cycle during regeneration of irradiated fuel and preparation of mixed uranium-plutonium fuel. This waste is distinguished by small quantities, very high specific activity, intensive heat release and high content of long-lived nuclides.

Waste management for the first category follows largely the traditional procedure (filtering, biofiltering, evaporation, sorption, concentrate solidification for liquid radwaste; sorting, pressing, burning for treated solid radwaste; compacting or long-term storage for untreated solid radwaste).

For the second category, waste management has no precedent either at Russian NPPs or at any foreign facilities.

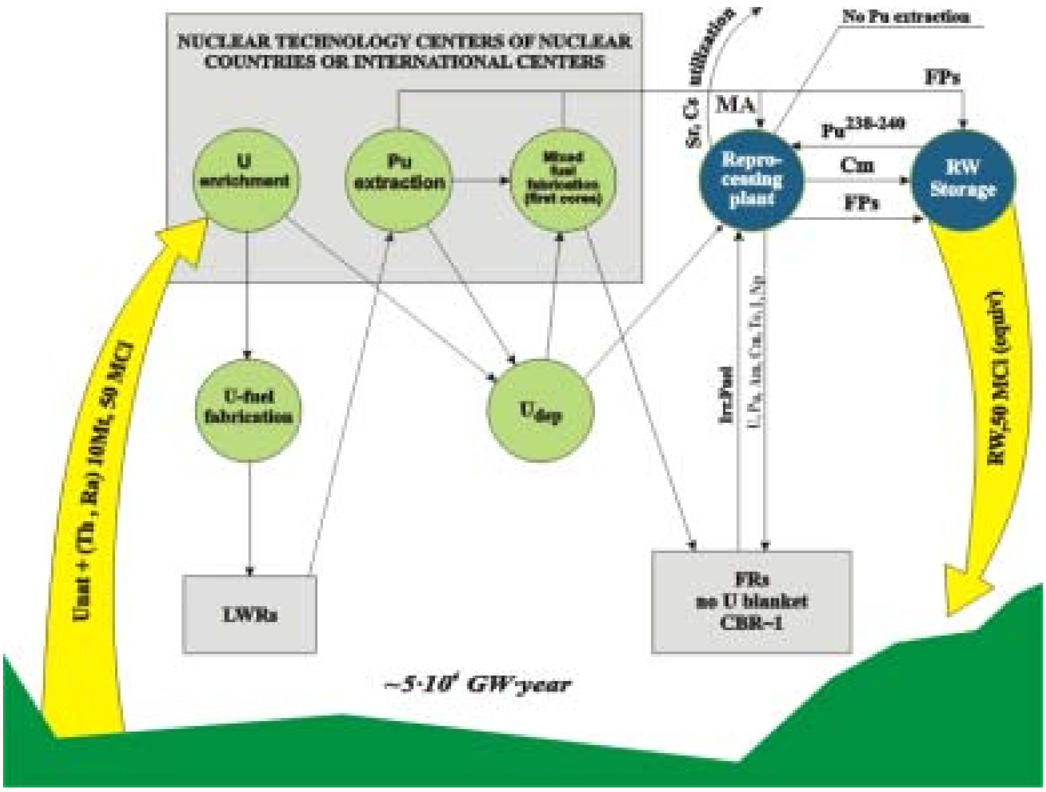


FIG. 3. Tentative scheme of radiation-balanced proliferation-resistant closed nuclear fuel cycle of large-scale nuclear power.

Opening of fuel rods and regeneration of irradiated nitride fuel give rise to high-level waste of the following categories:

— noble elements (ruthenium, rhodium, palladium, etc.) as well as molybdenum, zirconium, technetium present as particles in molten salts (electrolyte). Their total quantity makes 0.32 t per year. Separation is carried out with the use of a porous metal filter which is reconditioned by molten lead. When these fission products build up to 10 % by mass, they are removed into a container which after cooling, sealing and decontamination is sent on for long-term storage. Technetium is fractionated, with previous oxidation and distillation of the generated oxide Tc_2O_7. Technetium oxide undergoes condensation and will be stored before going to the reactor for transmutation;
— chlorides of rare-earth, alkali-earth and some other elements, mixed with electrolyte in the proportion of 1:1. Their total quantity is estimated at 1.8 t per year. Electrolyte regeneration involves previous extraction of uranium and plutonium, followed by separation of fission products through zonal crystallisation. The cleaned electrolyte is reused. The fraction of rare-earth elements and curium is enclosed in a nickel matrix with a fill of 10 % by mass. The alkali-earth metals fraction is enclosed in a copper matrix with a fill of 10 % by mass. Cesium chloride is placed in a calcium phosphate matrix with the same fill;
— fuel claddings and other structural components of fuel assemblies. Their total quantity is assessed at 3 t per year. The ingots together with spent crucibles are loaded into containers which, upon sealing and decontamination, are sent away for long-term storage. Consideration is being given to the possibility of recycling this metal in the on-site fuel cycle, after its treatment by induction melting;
— spent gas filters and gas absorbers. With their service life over, these components are to be loaded into containers and filled with matrix material (cement).

The radwaste management design provides for waste division into separate flows with regard to its activity, aggregative state and other characteristics, with subsequent treatment of each flow in the most efficient and safe way. The treatment results in transportable final products of minimised volume, which safely confine their radionuclides during transfer, storage and disposal.

The engineering design of the radwaste handling system in the on-site fuel cycle calls for further research and development work to validate the design solutions, especially those pertaining to regeneration, fractionation and treatment of high-level waste resulting from fuel regeneration and fabrication.

4. NEAR-TERM ACTIVITIES ON THE BREST-OD-300 FUEL CYCLE

i. Setting up a mock-up fuel cycle in NIIAR, Dimitrovgrad (2003-2004):
   — cutting of a spent fuel rod;
   — fuel reprocessing;
   — adjustment of fuel composition;
   — fabrication of nitride pellets;
   — fabrication of a new fuel rod.
ii. Improvement of the system of RW fractioning and storage to reduce the volume of storage facility intended for long-term storage of radioactive waste (2003 – 2005).
iii. Improvement of FC components based on trial operation (2003 – 2008).

5. REFERENCES

[1] White Book of Nuclear Power. General editing by Prof. E.O. Moscow, NIKIET (2001).
[2] LOPATKIN, A.V., et al., Radiation equivalency and nature emulation in radioactive waste management, Atomnaya energiya, **92**, No. 4 (2002), p. 308-317 (in Russian).

REVIEW OF EGYPTIAN ACTIVITIES IN UTILIZATION OF NUCLEAR ENERGY FOR ELECTRICITY GENERATION AND SEAWATER DESALINATION

M.A. EL SHRAKY, I.A. EL SHEHAWY

Nuclear Power Plants Authority
Cairo, Egypt

Abstract. Egypt has been considering for a number of years the utilization of nuclear energy to meet the combined challenge of increasing electricity and water demand on one hand and the limited primary energy and water resources on the other hand. In this regard, Nuclear Power Plants Authority (NPPA) was established to carry out the Egyptian nuclear power plants program and was entrusted with all aspects related to study, planning, design, and construction of nuclear power plants for electricity generation and seawater desalination. NPPA scope of activities includes site infrastructure development, survey and analysis of the most recent worldwide technologies relevant to advanced reactors and fuel cycles, preparation and evaluation of bid invitation specification, manpower development, as well as the necessary technical and economic feasibility studies. To achieve their objectives, NPPA has been carried out a number of integrated activities to provide the decision makers with detailed information regarding the feasibility and viability of the nuclear power option. This paper presents an overview of the current Egyptian activities in the field of cogeneration of electricity and potable water by nuclear energy.

1. INTRODUCTION

The developing of infrastructure of a country is a basic step towards improving its standards of living. Electric energy is an essential infrastructure element and a basic requirement for the social development and economic growth. Also, freshwater is essential for all terrestrial life. In arid and semi arid areas, the desalination of local saline water provides a considerable additional resource for supplying fresh water.

Because of the limited fossil fuel energy resources, the almost fully utilized hydro energy and the new and renewable energy can not produce enough energy to meet the demand growth, nuclear energy can play an important role in providing future energy, ensuring better energy mix and saving the ability of future generation to meet their own needs. Furthermore, in view of the unavoidable decline in the per capita share of the more or less constant natural fresh water resources in Egypt, seawater desalination is expected to play an increasing role in mitigating future deficit in potable water supply, particularly in remote desert areas.

With these constrains in mind, Egypt has been considering nuclear power as a suitable option to meet the country demands for electric energy and desalted water. A nuclear power plant providing the electricity to grid can in principle provide also electricity and/or heat to the desalination plants.

Following the oil crises in 1970s, Egypt considered an ambitious plan to build several nuclear power plants (NPPs). In order to implement this plan, it became necessary to set up an independent organization. Thus NPPA was established with the legal power and under the jurisprudence of the Minister of Electricity to prepare and implement the Egyptian NPPs program for electricity generation and seawater desalination.

During the early eighties, an extensive program to select and investigate the suitable nuclear sites was performed. El Dabaa site on the Mediterranean Sea coast has been chosen and qualified to construct the first Egyptian NPP.

Bid Invitation Specifications (BIS) to construct one or two pressurized water reactors of about 1000 MWe each were launched in 1983. The project was postponed in the wake of Chernobyl accident and the following strong resentment public opinion to nuclear power. The accident raised a big debate on the all levels and resulted in a common view towards the NPPs issue. The main output was to continue to complete the necessary infrastructure and to keep a state of readiness for efficient execution of the nuclear power program whenever an innovative nuclear reactor is developed. The required innovative reactor should essentially assure the highest degree of safety, low cost, minimal environment impact and could gain the public acceptance and confidence. The significant public concern centers on risk of the major accident and disposal of radioactive wastes. These concerns directly relative to the protection of the environment.

In preparation for new start, NPPA has carried out a number of integrated activities including the development of infrastructure of NPPs site at El-Dabaa and the necessary studies to provide the decision makers with detailed information regarding the prospects of using nuclear energy for simultaneous production of electricity and potable water. In that regard, several activities have been performed with the assistance of the International Atomic Energy Agency (IAEA), as will be shown below.

2. NUCLEAR POWER PLANTS SITE DEVELOPMENT [1, 2]

The site of El Dabaa Nuclear Power Plant is about 150 km west of Alexandria along the Egyptian Mediterranean Sea coast, Figure 1. The necessary site studies and investigations have been performed under the responsibility of NPPA, by the French consultant Sofratome. The field investigations and the final reports were revised and approved by the Swiss Consultant Motor-Columbus. Experts from the IAEA participated in a pilot experiment carried out in 1993 to define the atmospheric dispersion characteristics, to assess the impact of atmospheric release of gaseous effluent on the nearby populated area. The results of these works confirmed the suitability of the site to accommodate nuclear power units as well as a desalination plant.

A regional catalogue of historical and instrumentally recorded regional earthquakes till the year 1982 and later updated till 2000 have been established including the epicenter location, focal depth, intensity and magnitude. From these data and the tectonic structures survey, the reference earthquakes pertinent to the El Dabaa site have been defined.

El Dabaa Site is characterized by: low seismic activity, no recent or active faults, suitable foundation basement for construction of power block units and for the cooling water intake and discharge structures, suitable topography providing minimum constrains and good protection against sea hazards and low population density.

FIG. 1. El Dabaa nuclear power plants site.

The studies showed that the ecological impact of the NPPs is acceptable. The radioactive impact will be less than 10E-6 and 10E-3 of the admissible limits for liquid and gaseous normal releases respectively. In the case of accident release, the maximum dose could reach 3% of the admissible limits.

A study had been completed in mid eighties to evaluate the suitability of El Dabaa site to accommodate two nuclear units with a total capacity of 2000 MWe. This study showed that the site is suitable and the two units can be connected to the Unified Power System (UPS). Extending this work showed that no technical electrical problem would exist when the total installed capacity is increased.

During the last 20 years, meteorological and marine data have been continuously processed from special detectors and a meteorological station tower at the site. Also, NPPA has succeeded in developing the site infrastructure such as water, electricity, road networks, sanitary system, and transportation according the latest techniques and standards in each field. Furthermore, NPPA is developing El Dabaa site infrastructure master plan and started implementation of the plan.

3. FEASIBILITY STUDY OF NUCLEAR POWER AND DESALINATION PLANT ON EL DABAA SITE

In 1998, NPPA started study with IAEA technical assistance to investigate the feasibility of nuclear power and desalination plant on El Dabaa site. The main objective of the study was to provide the decision-makers with the necessary information regarding the technical and economical feasibility and viability of the nuclear option for electricity generation and seawater desalination. The scope of the study covered in depth the following fields:

(i) Quantification of future needs of electricity and water based on socio-economic analysis of the Egyptian situation to determine the likely development scenarios and their impact on future demands.
(ii) Identification of available technical options through state-of-the-art review of both desalination technologies and commercially available nuclear power plants.
(iii) Assessment of local participation capabilities and manpower availability.
(iv) Cost comparison of various nuclear and conventional power production and seawater desalination options as well as assessment of financing options.

The main findings of the feasibility study are outlined below.

3.1. Electrical system analysis

To develop a national energy policy, a survey is needed of the country‘s available and potential energy resources.

3.1.1. Status of electrical system [3, 4]

The first power station was built in 1932 in Cairo with installed capacity of 20 MWe. Thermal power plants, mainly steam, continued to be constructed. In 1960, the hydro plants were introduced with the commissioning of Aswan Dam hydro station with 345 MWe installed capacity. The High Dam hydro plant was completed in 1970 with a total installed capacity of 2100 MWe. By introducing the two main hydro power plants their installed capacity accounted for about 65 percent of the total installed capacity of 3775 MWe in 1970.

During the period 1970-2000, an ambitious program of power generation expansion was launched. More than 10 GWe capacities were added to the electric system. This installed capacity consisted of some gas turbine and steam units using either heavy fuel or natural gas or both. It is worth mentioning that the last decade was characterized by the introduction and utilization of combined cycle generating plants. The first wind station was commissioned in 2000 with 63 MWe installed capacity. Fig. 2 represents the development of electricity generation in Egypt in the period 1970-2000.

The statistics for the year 2001/2002 indicated that the generating system capacity in Egypt was about 16.6 GWe and the energy generated was about 80.3 TWh. The peak load was about 13.3 GWe and the yearly electrical energy growth was 7.7%. The installed capacities of steam, combined cycle, gas turbine, hydro and wind units were 9.9 GWe, 2.6 GWe, 0.7 GWe , 2.7 GWe and 63 MWe respectively. It is worth to mention that the hydro share in the generated energy is constrained by the irrigation regime and the ambient temperature for combustion turbines and the ageing for some units influence the net available capacity.

The Unified Power System was commissioned in 1967 and connects all the power station in Egypt (except the remoted areas). The current largest hydro unit is 175 MWe and the largest steam unit is 627 MWe.

3.1.2. Projected electricity demand

The energy and load demand forecast is an essential component of the operational and the planning management of the power system. The basic objective of an electric power generating expansion plan is to provide sufficient power to meet the demand taking into account the different aspects related to power system reserve requirements and economics.

The load forecast modeling for the Egyptian power system is the basis upon which many of the calculations and decisions pertaining the load forecast is made. The forecast has been developed using the econometric modeling and regression analysis.

Using the available database of the electricity sales and prices for different sectors, three scenarios were derived. The medium scenario of electric energy demand indicates that:

(i) The electrical energy will increase rapidly to meet the sustainable development in the next 20 years; the generated energy in year 2020 will be three times the energy generated in year 2000, the average rate of increase is about 5.5% / year.

(ii) The peak load will increase also with the same rate as the energy, the expected peak load in year 2020 will be about 35 GWe.

(iii) The new capacity needed to meet the load demand is about 1000 MWe / year, this increment of addition will be increased when taking into consideration the new capacity needed to substitute the retired units.

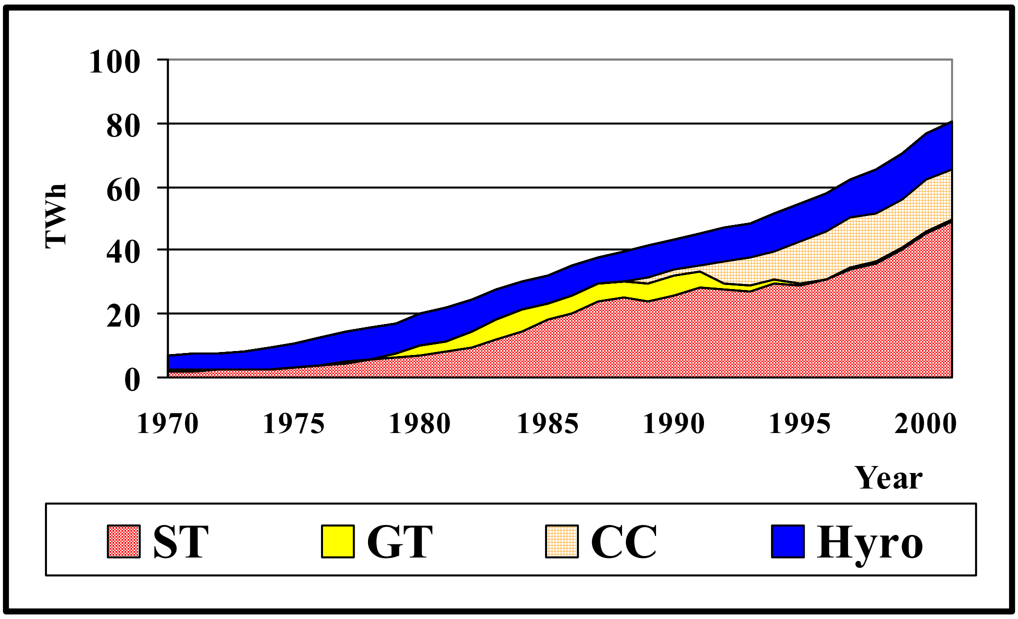


FIG. 2. Development of electricity generation in Egypt in the period 1970-2000.

3.2. *Water system analysis [3, 5, 6]*

3.2.1. Status of water resources

The uneven distribution of water on earth has produced a large expanse of arid and semi arid lands in the middle latitude, including Egypt. In arid and semi arid areas, the desalination of local saline water supplies holds considerable promise for supplementing fresh water. The other alternative is the transportation of fresh water over long distances. This is an economic problem involving supply of freshwater particularly to the remote areas. Desalination technologies are investment-intensive and, generally, economic on large scale. Therefore, water resource analysis as well as the projected future water demand, supply and deficit are vital components in determining the feasibility of saline water desalination.

Natural water resources in Egypt consist of renewable and non-renewable resources. The Nile water is the main renewable resource. The non-renewable resources include fossil groundwater. In addition to these resources, there are unconventional sources consisting of Nile Valley aquifers, reuse of treated agricultural drainage water, industrial waste water and sewage water. These unconventional sources cannot be considered as resources in themselves and cannot be added to Egypt's share of fresh water.

As mentioned above, the Nile water is the most significant surface water source in Egypt. The Nile water division agreement of 1959 allocates 55.5 billion m^3 annually to Egypt. The per capita share that was more than 2100 m^3/year in 1960, came under the scarcity line of 1000 m^3/ year in 1995 and continues to decline as indicated in Figure 3.

3.2.2. Total projected demand

There are four main consuming sector of fresh water in Egypt. These are irrigation, municipal, industrial and navigation. A comparison between projected water resources and demand is shown in Figure 4. The Figure shows that the total water resources requirements will increase by about 5% than the projected in the year 2020. The analysis of the expected freshwater supply indicates that as of 2012 a deficit in freshwater would develop and will reach about 6 billion m^3 within 20 years.

It is expected that part of the future deficit will be covered through improving water management regimes, reduction of distribution losses and new water resources including desalination of brackish water.

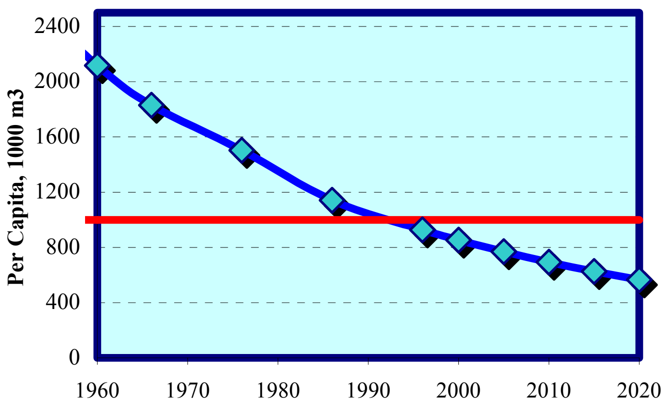


FIG. 3. Per capita share of natural freshwater.

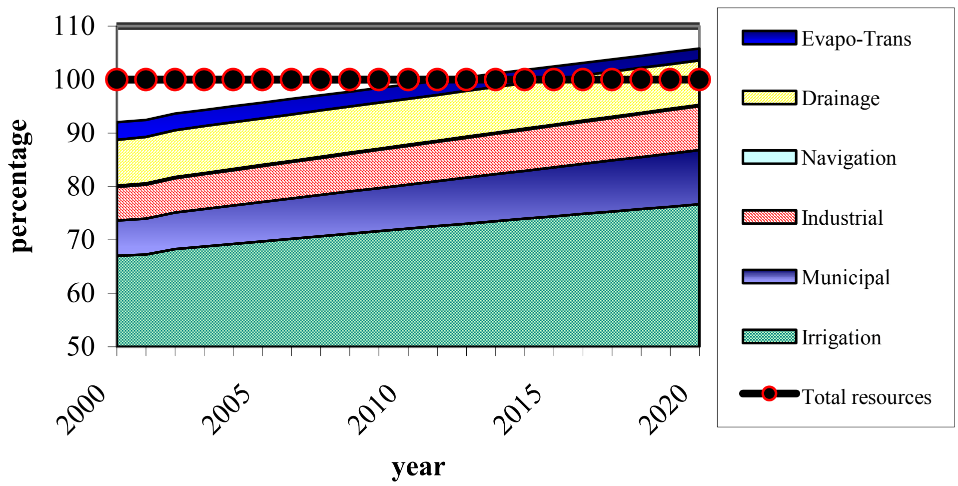


FIG. 4. A comparison between projected water resources and demand.

3.2.3. Desalination

Desalination technology was introduced into Egypt as early as 1926 when a land marine-type desalination plant was built in an oil field. In the past 30 years desalination plants of various sizes and technologies have been introduced to Egypt as a solution to the problem of limited natural fresh water resources in remote and isolated areas, as well as for industrial applications. A recent survey by the International Desalination Association IDA indicated that the total desalted water inventory in Egypt increased from about 2000 m^3/d in 1972 to more than 192,000 m^3/d at the end of 2000 as can be seen in Figure 5.

It is expected that the demand on desalination technologies will continue to increase in all sectors in Egypt. Part of the future needs would be to meet the demand of the consuming sectors. The other part would be to cover deficit in freshwater supply.

Based on the extrapolation of the historical data and assuming the desalination plant life is 15 years and the seawater desalination will cover only 10% of the deficit in the potable water supply, the future requirement of the seawater desalination capacities is shown in Table I. The Table indicates the total capacity would be more than 1.9 millions m^3/day in 2020. Some of the future capacity in the years 2012 and beyond is expected to be sufficiently large in magnitude that it will support the installation of seawater desalination production facilities larger than 100,000 m^3/d. The analysis indicates that the size of the proposed the nuclear desalination plant for El-Dabaa site would be about 150,000 m^3/d.

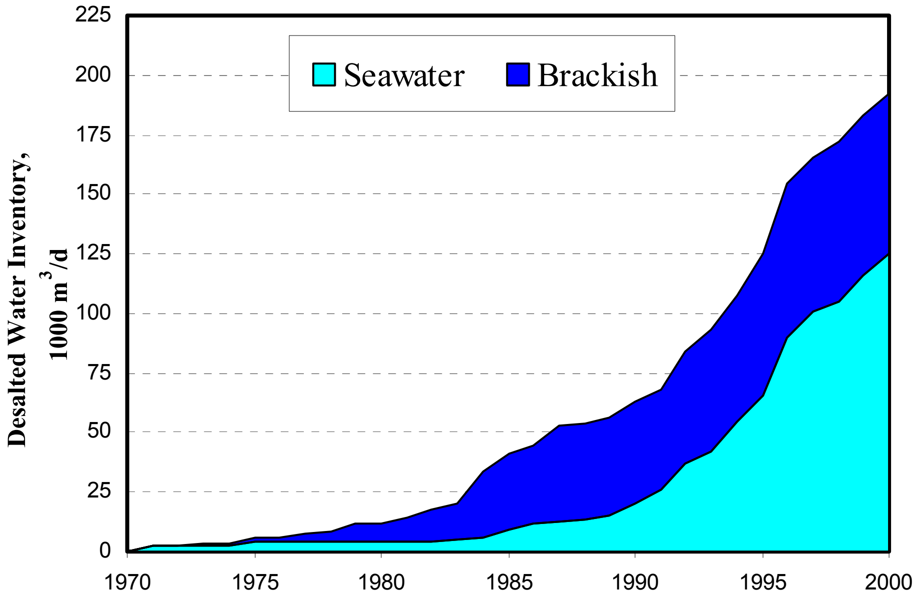


FIG. 5. Desalted water inventory in Egypt in the period 1970 – 2000.

3.3. Cost estimates [6, 7]

The study estimated the cost of power production at El Dabaa under varying assumptions and examined the economic merits of the El Dabaa Nuclear Power Plant by comparing it with alternative power plants, which would perform the same services.

The performance and cost analysis was carried out using the Desalination Economic Evaluation Program (DEEP) and another spreadsheet developed in NPPA to calculate the levelized energy generating cost and to perform a cash flow analysis. The ground rules of the economic estimate was based on: the starting operating date 2010, the reference date January 2000 and without escalation, the economic lifetimes 40 years for the nuclear and 35 for conventional and the interest/discount rate 8%. The estimated relative levelized energy generating cost for pressurized water reactor of 1000 MWe and 600 MWe, the conventional steam plants of 325 MWe and 650 MWe as well as combined cycle of 500 MWe is shown in Figure 6.

Table I. Projections of future seawater desalination inventory, 1000 m^3/day

YEAR	Expansion Requirements of Current Uses		Required Desalted Capacity	Total Desalted Capacity
	Total Contracted Capacity	Actual Operating Capacity		
2000	129	112	0	112
2005	202	165	0	165
2010	293	219	0	219
2015	401	272	690	962
2020	527	326	1600	1926

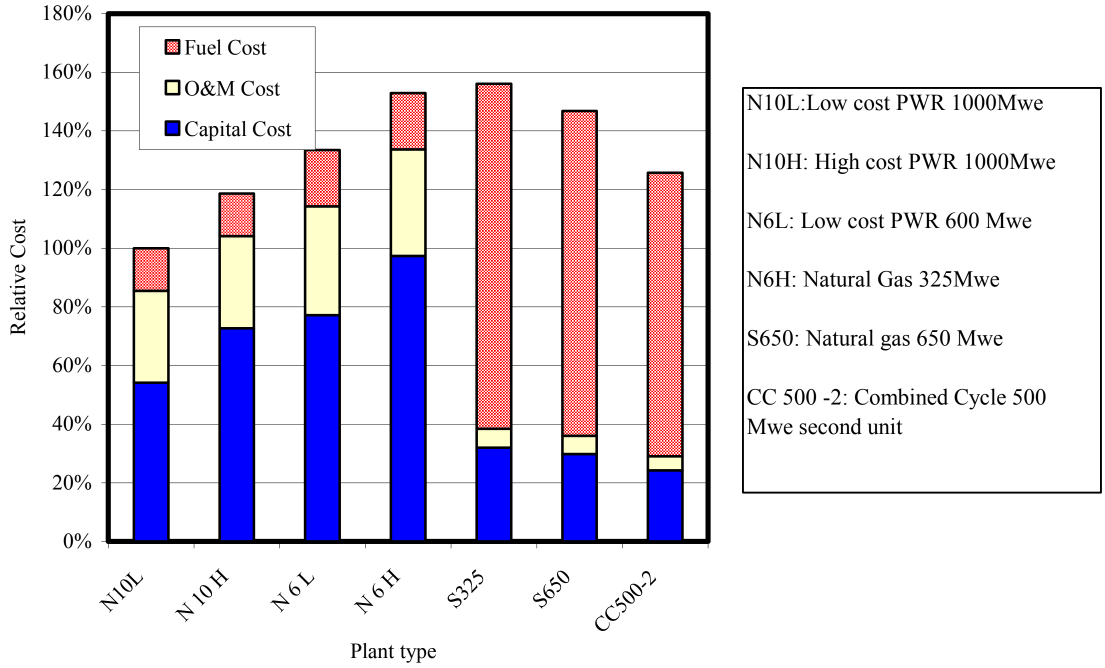


FIG. 6. Levelized electric generation cost.

Sensitivity analysis was performed in order to test the impact of variations of the important parameters affecting the generating costs. The relative impact on the lowest generating cost of nuclear and conventional plants due to the variation of annual operating hours per year, discount rate, and fuel costs are shown in Figures. 7, 8 and 9 respectively.

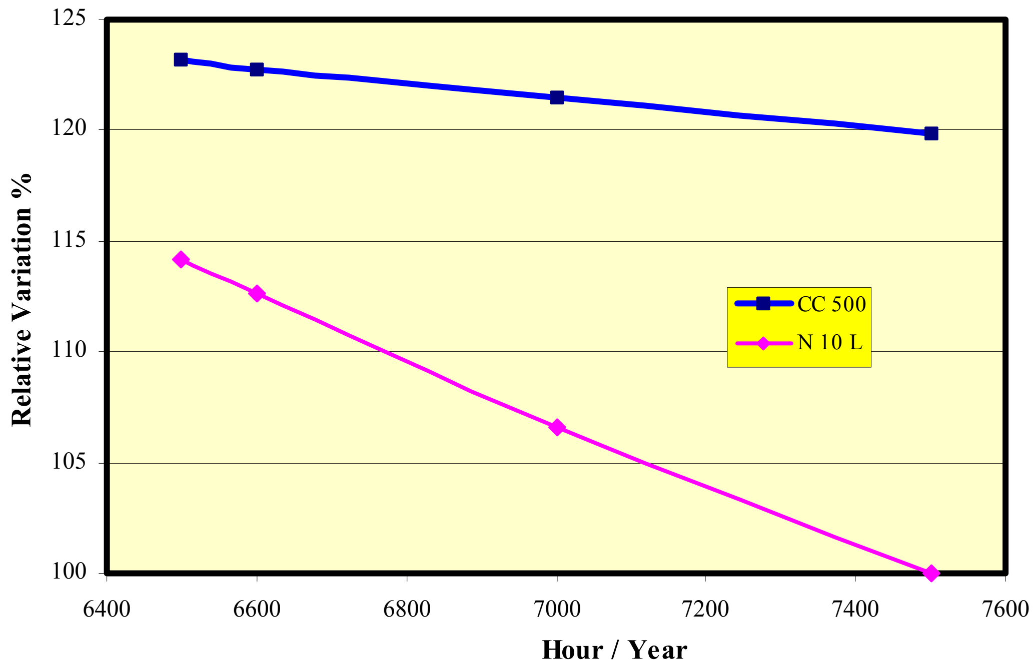


FIG. 7. Variation of annual operating hours.

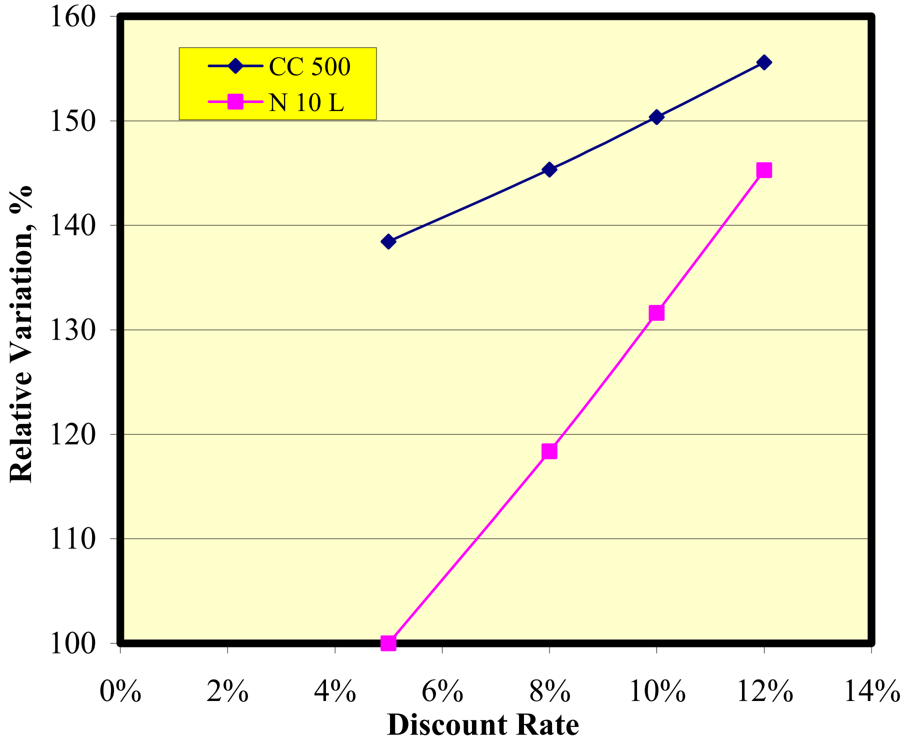


FIG. 8. Variation of discount rate.

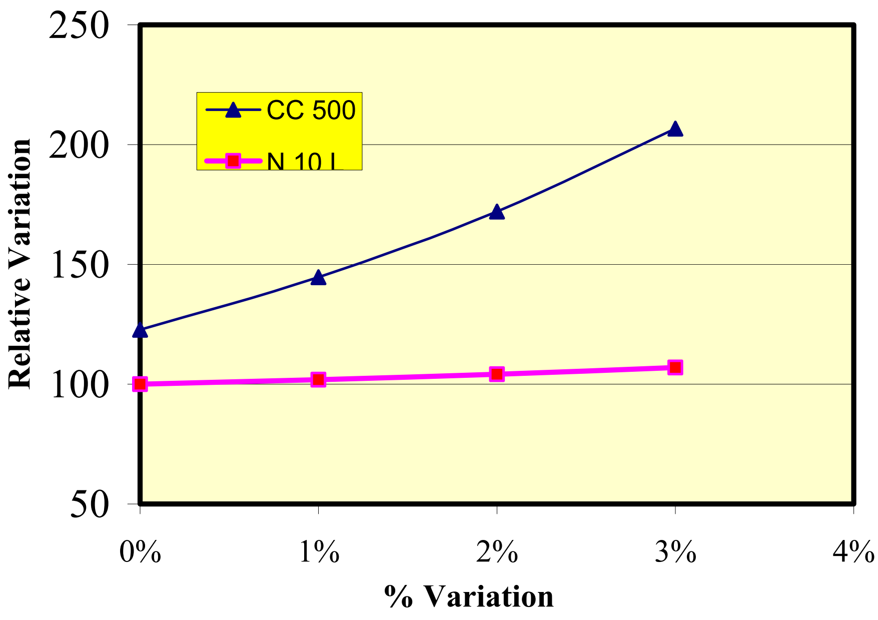


FIG. 9. Fuel cost variatio.

3.4. *Financial requirement*

A turnkey contractual approach will be adopted for the first NPP project with maximizing the local participation and manpower participation in all stages of the project. In addition, long term loans with prevailage interest rate and repayment period are required. These types of loans are needed to finance the major part of the total foreign components and the remaining will be covered from national resources.

4. TECHNICAL COOPERATION PROJECTS (TCPS)

In the framework of the activities are being carried out by NPPA and with the aim to maximize the benefits of international cooperation, NPPA is currently cooperating with IAEA in implementing numbers of TCPs. These cooperative works are complementing the national activities and aiming to receive the technical support and advice in developing the infrastructure of NPPs site, improving the human resources capabilities and helping in the planning and implementing the nuclear power project for electricity generation and seawater desalination. The main objectives and scope of the main TCPs are outlined in the following.

4.1. *Simulation of nuclear desalination plant*

El Dabaa NPPs site is equipped with a basic simulator for training purpose. It simulates the basic principle operation of water cooled pressurized reactors as well as the accidents might be occurred in the pressurized light water reactor. The objectives and scope of this TCP are to investigate El Dabaa basic simulator and identify potentials for upgrading and to develop the necessary desalination mathematical model to incorporate into the simulator.

4.2. *Experimental program for investigation of feedwater preheating on the performance of reverse osmosis (RO) membranes*

The main short–term objective of the project is to study performance characteristics of three commercial seawater water RO membranes over the range of allowed temperatures and pressures. The long-term is to study the effect of continuous operation at maximum feedwater temperature and pressure on RO membrane life. The Experimental Facility is constructed at El Dabaa site and the designated tests will be performed soon.

4.3. *Development of an integrated economic and financial assessment tool for power/ desalination systems.*

The project aims to increase the computational capability and flexibility of DEEP computer code and to use it as an economic and financial tool for the application of the national case studies. The scope covers development and incorporation of water transport cost module, additional module for financial analysis, and a data base module for the power and desalination models.

4.4. *Human resources development for nuclear power plant project preparation and project management*

The main objectives of the project are to transfer knowledge, information and experience related to the development of human resources for planning and implementing a nuclear power project for electricity generation and seawater desalination. The scope of the project includes identification of tasks and resources required for implementation of Dabaa NPP project, define the site development master plan requirements, and preparation of terms of reference for consulting services and the bid invitation specifications for the first NPP.

5. CONCLUSIONS

It is clear that installing a nuclear power plant in Egypt, at El Dabaa site, to produce electricity and potable water, has potential to bring major social and economic benefits. Also, it is obviously there is a need for further activities involving development and upgrading to support nuclear option. This will require substantial efforts on both national and international levels. Developing an innovative reactor design is technically feasible, economically convenient, financially viable and environmentally clean will gain the public acceptance and promote utilization of nuclear energy in Egypt.

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STUDY ON REPROCESSING METHOD OF HIGH TEMPERATURE GAS COOLED REACTOR FUELS

S. KATANISHI, M. TAKEI, K. KUNITOMI

Japan Atomic Energy Research Institute
Oarai-machi, Ibaraki-ken, Japan

Abstract. In high temperature gas-cooled reactor (HTGR) fuel elements, coated fuel particles which consist of UO_2 fuel kernels with TRISO coatings are dispersed in a graphite matrix. Since the reprocessing of UO_2 fuel kernels are basically same as that of light water reactor (LWR) fuels, that is purex process, additional process in order to supply HTGR fuel into reprocessing plant for LWR fuels was devised as head-end process. The 1st step of head-end process is to remove the matrix graphite and carbon outer coating layer of coated fuel particles. The 2nd step is to remove the SiC layer of coated fuel particles. In the 3rd step, two inner graphite layers are roasted and removed from UO_2 fuel kernels. UO_2 is oxidized and U_3O_8 is obtained by this step. In conditioning process after head-end process, U_3O_8 is dissolved with nitric acid and diluted with depleted uranium solution in order to make acceptable condition of purex proccess of reprocessing plant for LWR fuels in Japan. Conceptual design of head-end process facilities and estimation of cost for this process was discussed. And feasibility was certified technically and economically even if the reprocessing option is adopted according to national policy of Japan.

1. INTRODUCTION

High temperature gas-cooled reactor (HTGR) fuels are expected to be a suitable for once through with final disposal after interim storage because of its high burn-up and less amount of residual fissile materials. However, since the national policy of nuclear energy in Japan is to establish a fuel cycle with reprocessing of spent nuclear fuels, feasibility of reprocessing of HTGR fuels was investigated.

Japan Atomic Energy Research Institute (JAERI) has started a design study of Gas Turbine High Temperature Reactor with electric power of approximately 300MW (GTHTR300)[1]. Design and operational experience obtained from the High Temperature Engineering Test Reactor (HTTR)[2] of JAERI are applied to this study. The goal of this study is to establish the design of the system with high safety characteristics and high economical advantages. Through this design study, a specification of HTGR fuels, and a property and a quantity of spent fuels of HTGRs have been clarified. Technical feasibility and economical feasibility of reprocessing in case of the GTHTR300 fuels were investigated.

After UO_2 fuel kernels are taken out from fuel elements, the same reprocessing process as that for LWR fuels must be applied. An idea that head-end process facilities for HTGR fuels are built in the adjacent place to reprocessing plant for LWR and that UO_2 fuels taken out from the spent fuel elements are supplied into the reprocessing process was devised. Reprocessing plant for LWR fuels (Rokkayo reprocessing plant) is now under construction in Rokkasyo-mura village by Japan Nuclear Fuel Limited. The each stage which is necessary in head-end process and conditioning process in order to supply the solution of uranium into reprocessing process were investigated. Outline of the facilities in the head-end process, such as specification, capability and numbers of facilities, were decided. Cost for construction and operation of the facilities was also estimated.

JAERI has been entrusted with this study from Ministry of Education, Culture, Sports, Science and Technology of Japan.

2. SPECIFICATION OF GTHTR300 FUELS

2.1. Structure of fuels

Schematic of fuel structure for HTGRs are shown in Figure 1. A coated fuel particle (CFP) consists of a UO_2 fuel kernel with 4 layers of coatings. The inner coating is a low density, porous pyrolytic carbon (PyC) buffer layer, that is a plenum of gaseous FPs and protects the 2nd layer from swelling of the fuel kernel. The 2nd layer is an inner PyC (IPyC) layer made of isotropic PyC, that is a barrier for the diffusion of gaseous FPs and protects the interaction between the 3rd layer and a fuel kernel. The 3rd layer is a silicon carbide (SiC) layer, that is a structural components and a barrier for the diffusion of metal FPs. The 4th layer is outer PyC (OPyC) layer as a barrier for the diffusion of gaseous FPs and a protection of the SiC layer. In a fuel compact, CFPs are dispersed in a hollow cylindrical graphite matrix. A fuel rod is composed of fuel compacts which are piled up and supported by center rods in the coolant channel of graphite fuel block in reactor core, as shown in Fig. 1. Major dimensions of the GTHTR300 fuel are shown in Table I.

For reprocessing of HTGR fuels, a head-end process to take out CFPs from fuel compacts and to take out UO_2 fuel kernels from CFPs is necessary due to these structural features.

Table I. Dimension of fuels

Coated fuel particle	
Fuel kernel diameter	550 (μm)
Buffer layer thickness	140 (μm)
IPyC layer thickness	25 (μm)
SiC layer thickness	40 (μm)
OPyC layer thickness	25 (μm)
Outer diameter of coated fuel particle	1010 (μm)
Fuel compact	
Length	83 (mm)
Outer diameter	26 (mm)
Inner diameter	9 (mm)
Fuel rod length	1050 (mm)

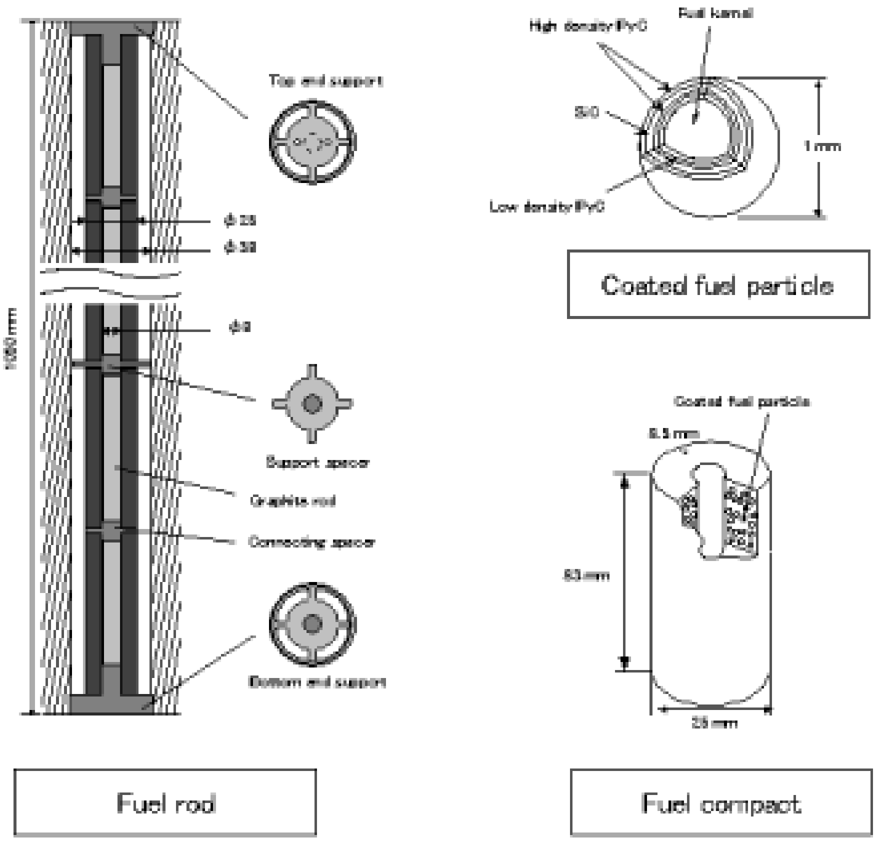

FIG. 1. Structure of HTGR fuel.

2.2. Features of spent fuels

Property of spent fuels in HTGRs and light water reactors (LWRs) were calculated by using ORIGEN code[3, 4]. Calculation conditions are listed in Table II.

Calculated residual enrichment of U-235 and isotopic composition of Pu-240 are listed in Table III. Calculated radioactivity and decay heat of HTGR fuels are two to tree times larger than these of LWR fuels, as shown in Figure 2. The neutron emission and the gamma emission were calculated to be the same level as these of LWR fuels.

Table II. Calculation conditions

	GTHTR300	ABWR	PWR
Power density (MW/t)	82	26	40
Burn-up (GWd/t)	120	45	45
Initial enrichment (%)	14.0	3.8	4.1
Calculation code	ORIGEN-S	ORIGEN2.1	ORIGEN2.1
Library	HTGR	BWR-U	PWR-U50

Table III. Calculated property of HTGR spent fuel as discharged

	GTHTR300
Enrichment of U-235	4.5%
Isotopic composition of Pu-240	17.4%

3. DISCUSSION ON HEAD-END PROCESS

The each stage which is necessary in head-end process and conditioning process in order to supply the uranium solution into reprocessing process were investigated.

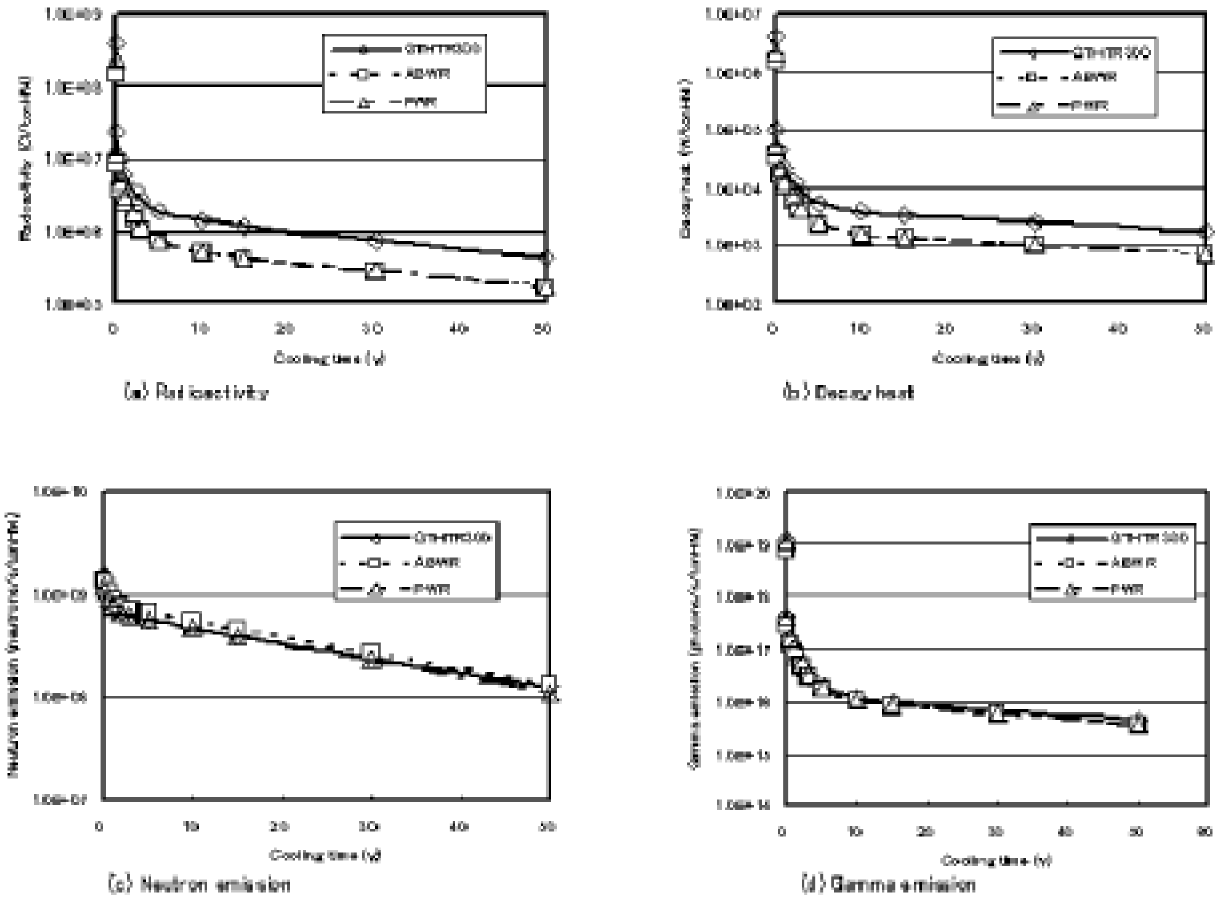

FIG. 2. Calculated properties of HTGR fuel.

3.1. Process to take out coated fuel particles from fuel compact

In the fuel compact, CFPs are dispersed into graphite matrix. The 1st step of head-end process is to take out the CFPs from fuel compacts. Some methodologies, such as electrical method, chemical method, mechanically crushing method and combustion method, were already proposed for this step. Complete separation of CFPs and graphite is not possible by electric method. The chemical method has problems such as the low separation efficiency, difficulties of recovery of chemicals and corrosion of materials. Recovery rate of graphite and fuel particles are low in the mechanically crushing method. After consideration of these features of each method, combustion method with burning of graphite under air environment was selected as a reference method for present investigation, because this method has been already adopted in the recovery of faulty CFPs in the production plant of HTTR fuels though it was small scale.

3.2. Process to take out fuel kernels from coated fuel particles

After removal of graphite by burning in the 1st step, the CFPs with SiC layer are obtained. The SiC layer is a structural component of CFPs and is chemically stable. The 2nd step of head-end process is to take out the fuel kernels from the stable SiC layer. Some mechanical method for crushing the SiC layer, such as roller method, jet mill method and rotating dist method, were already proposed. Roller method has a problem on keeping of gap width to crush the SiC layer. Maintenance ability and confinement of radioactive materials are also problems in this method. In the jet mill method, CFPs are accelerated in a fluidized bed and collide with a plate to crush the SiC layer. The facility for separation of small fragments of SiC from exhaust gas is necessary in this method. The rotating disk method is to crush the SiC in the gap between static disk and rotating disk. This method has been adopted in the recovery of faulty CFPs in the production of HTTR fuels and the availability was established. The rotating disk method was selected as a reference method for present investigation.

3.3. Process to remove inner graphite layer from coated fuel particles

Fuel particle after crushing the SiC layer in the 2nd step has IPyC layer and buffer layer. The 3rd step is to remove these two graphite layers by roasting in the air environment. Mixture of oxidized fuel (U_3O_8) and small fragments of SiC is obtained after the 3rd step.

4. DISCUSSION ON CONDITIONING PROCESS

4.1. Requirement on conditioning of solution

HTGR fuels after head-end process is supplied into separation process of the reprocessing plant for LWR fuels. The solution of spent fuel must meet the acceptable conditions of separation process of Rokkasyo reprocessing plant which are shown in Table IV [5]. These are conditions in order to ensure the criticality safety. Sampling and analysis of solution are conducted at the upstream of separation process and the solution is supplied into separation process after certifying that the conditions of solution meet the acceptable conditions.

Table IV. Acceptable condition of separation process for criticality safety

	Limitation
Enrichment of U-235	less than 1.6wt%
Isotopic composition of Pu-240	larger than 17w%
Concentration of uranium	less than 300gU/l
Concentration of plutonium	less than 3.5gPu/l

Enrichment of U-235 of HTGR spent fuel was calculated to be 4.5%, as shown in Table III. This value does not meet the acceptable condition. In case when the concentration of uranium in the solution of spent fuel of HTGRs is adjusted to be 250gU/l, concentration of plutonium becomes 6.6gPu/l. This value is also beyond the limitation.

In order to meet the acceptable condition, a process for conditioning of solution of spent fuel is necessary after the head-end process. In this conditioning process, the U_3O_8 is dissolved with nitric acid and fragments of SiC were removed. The nitric solution of spent fuel is diluted with depleted uranium solution in order to meet the condition for enrichment of U-235. When the solution is diluted to 3.5 times with depleted uranium solution with enrichment of 0.2wt%, enrichment of U-235 becomes 1.43wt% and concentration of Pu becomes 1.43wt%. This diluted solution meets the acceptable conditions.

On the other hand, isotopic composition of Pu-240 is calculated to be 17.4%, as shown in Table III. This calculated value meets the acceptable condition. However, it will be possible to have smaller value due to the dispersion of burn-up distribution in each batch of fuels during

reactor operation. In order to solve this problem, the solution of HTGR fuel is mixed with solution of LWR fuel which has higher composition of Pu-240 than 17% in the measurement and adjustment tank of reprocessing plant.

4.2. Process flow

Process flow of conditioning process was discussed. Fig. 3 shows the outline of the process. HTGR spent fuels (mixture of U_3O_8 and fragments of SiC) after head-end process is accepted by spent fuel supply hopper from head-end process and supplied into dissolver tank to be dissolved by nitric acid. SiC fragments are left as undissolved residue and removed by screw conveyor to washing tank. Nitric solution of spent fuel is moved to solution catch tank and supplied to centrifugal clarifier to remove the undissolved residue. Clarified solution is supply to mixing tank. Then, residual enrichment of U-235 and concentration of uranium are adjusted by mixing with solution of depleted uranium which is supplied from depleted uranium process. Adjusted solution of spent fuel is supplied to separation process of Rokkasyo reprocessing plant after analysis and certificate of condition.

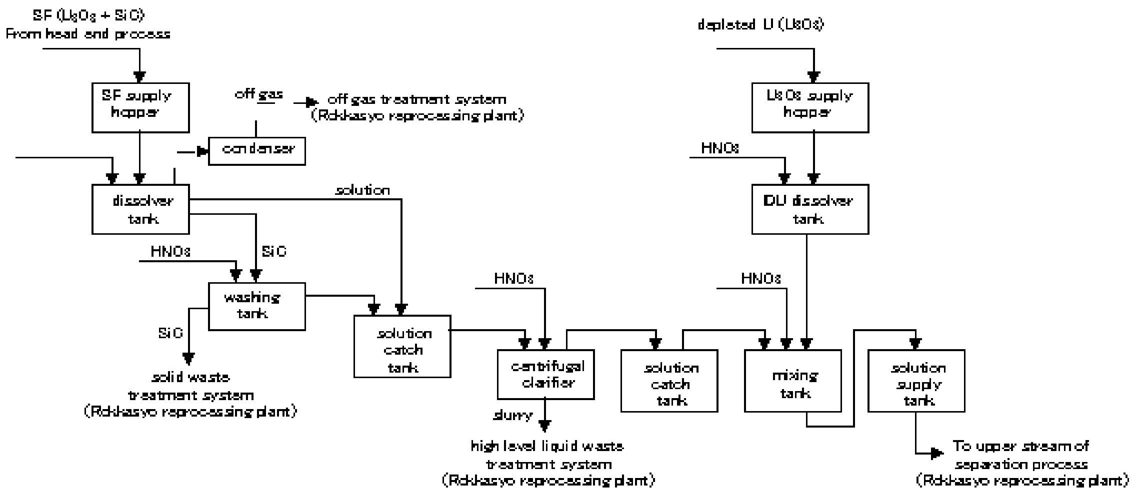

FIG.3. Process flow of conditioning process.

5. DISCUSSION ON SAFETY OF REPROCESSING PLANT

One of the most important safety items of reprocessing plant is criticality safety. However, since geometry and dimension control are conducted at the major process after the separation process in Rokkasyo plant, there will be no any problem in the reprocessing of HTGR fuels if the acceptable condition is attained.

In separation process and waste treatment process, radioactivity and decay heat in reprocessing of HTGR fuels are approximately 30% larger than these for LWR fuels. And they are 20% larger in process to treat Pu. This problem can be solved by prolongation of cooling period of fuel up to 4 years, that is the same as cooling period of LWR fuels. Also, mixing with solution of LWR fuels in the measurement and adjustment tank is another way to solve this problem.

6. DISCUSSION ON COST OF REPROCESSING

6.1. Facilities of head-end process and conditioning process

Outline of facilities of head-end process and conditioning process was designed in order to estimate the cost. Arrangement of facilities is shown in Figure 4.

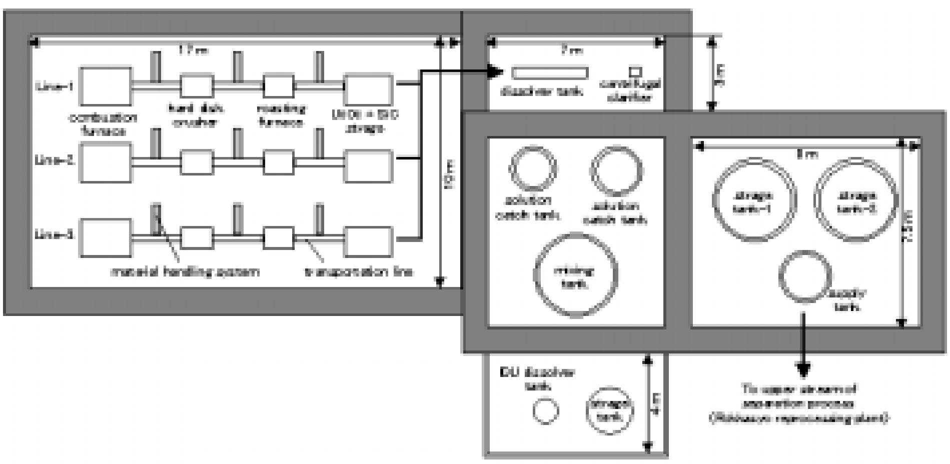

FIG. 4. Arrangement of facilities of head-end process and conditioning process.

6.1.1. Head-end process

Major facilities of head-end process are:

(i) combustion furnace for burning of graphite of fuel compact,
(ii) hard disk crusher for crush the SiC layer of CFPs with static and rotating disks,
(iii) roasting furnace for roasting of CFPs with graphite layers.

Amount of uranium in spent fuel of HTGRs is assumed to be 31ton in a year. This means that heavy metal (HM) of 130kg must be reprocessed in a day. Design of facilities with capability of 10kg-HM/batch and 4batch/day are considered to be possible from the experience of recovery system of faulty fuel in HTTR fuel production. It leads that 3 lines of above-mentioned facilities are necessary in head-end process. They are arranged in a building as shown in Fig.4. And facilities for adsorption of gases from combustion of fuel compact, SiC crushing and roasting of CFPs are necessary. The distance between each facilities is more than 1m in order to keep criticality safety.

6.1.2. Conditioning process

Spent fuel supplied from head-end process is mixture of U_3O_8 and fragments of SiC. This mixture is supplied into dissolver tank to dissolve U_3O_8 with nitric acid and to remove SiC fragments. Schematic of dissolver tank is shown in Figure 5.

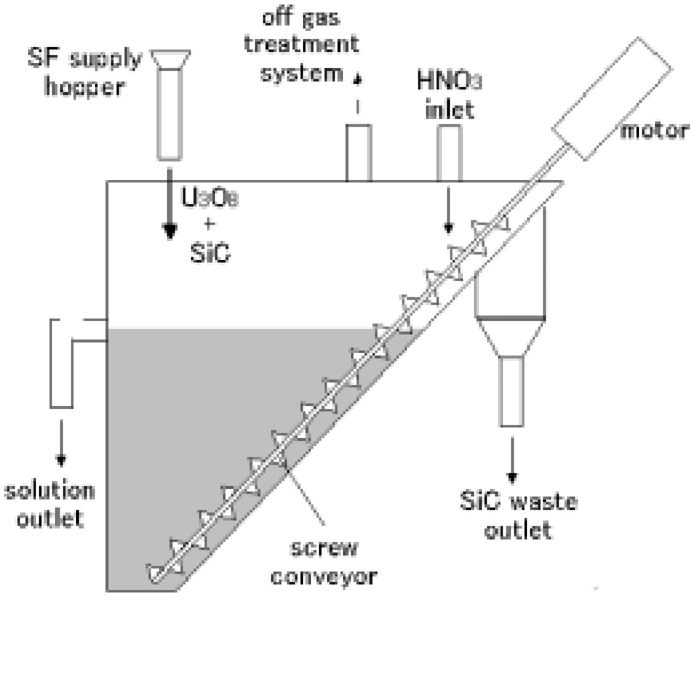

FIG. 5. Schematic of dissolver tank in conditioning process.

The spent fuel is supplied into solvent in dissolver tank from the top by spent fuel supply hopper and reaches the bottom end of screw conveyor. Fuel is drawn up by the screw conveyor and dissolved with nitric acid. The solution is drained at overflow outlet line. SiC fragments are also drawn up to beyond the solvent level by screw conveyor and washed by nitric acid which is supplied from HNO_3 inlet. They are discharged at the SiC waste outlet from dissolver tank.

Major components of conditioning process are listed in Table V. Geometry and dimension of each component were determined in the point of view of geometry control for critical safety, margin for corrosion and manufacturing error. Conditioning process area in the building is divided into 4 compartments for maintenance ability as shown in Fig. 4. The distance between each facility is more than 1m in order to keep criticality safety.

Table V. List of components of conditioning process

Components	Number	Type
Spent fuel supply hopper	1	Vertical cylinder
Dissolver tank	1	Flat plate
Solution cooler	1	Double tube
Solution catch tack	1	Annular type
Condenser	1	Multi pipe
SiC washing tank	1	Flat plate
Centrifugal clarifier	1	Bowl type
Clarified solution catch tank	1	Annular type
Mixing tank	1	Annular type
Supply tank	1	Vertical cylinder
Storage tank	2	Vertical cylinder
Depleted U supply hopper	1	Vertical cylinder
Depleted U dissolver tank	1	Vertical cylinder
Depleted UNH storage tank	1	Vertical cylinder

6.2. Cost of reprocessing

Construction cost of building and facilities, personnel cost and cost for material and article for consumption were estimated based on the outline design of facilities. Construction cost of building and facilities in a year was estimated with the assumption that the life of building and facilities is 30 years. It is also assumed that depleted uranium could be obtained with no expense. Estimation results show that the cost for head-end process and conditioning process is 27,000yen/kg-HM. In case of reprocessing for HTGR fuels, this cost must be added to the reprocessing cost for LWR fuels, since these processes are additional process to that of LWR fuels. However, total reprocessing cost in HTGR fuel per unit electricity generation is 0.18yen/kWh while it is 0.63yen/kWh in LWR fuels [6]. This difference results from high plant efficiency and small quantity of spent fuel generation in HTGRs.

Total fuel cost including front-end and back-end of fuel cycle was estimated to be 1.32yen/kWh in HTGRs while it was 1.65yen/kWh in LWRs [6], as shown in Fig. 6. However, it must be noticed that this total cost in HTGRs is not including cost for disposal of irradiated graphite materials which is generated at every refueling work. The treatment and disposal method of these graphite materials will be investigated and the cost will be certified in future. However, it can be concluded that the total fuel cost of HTGRs possibly be the same level as that of LWRs even when the graphite treatment cost is included.

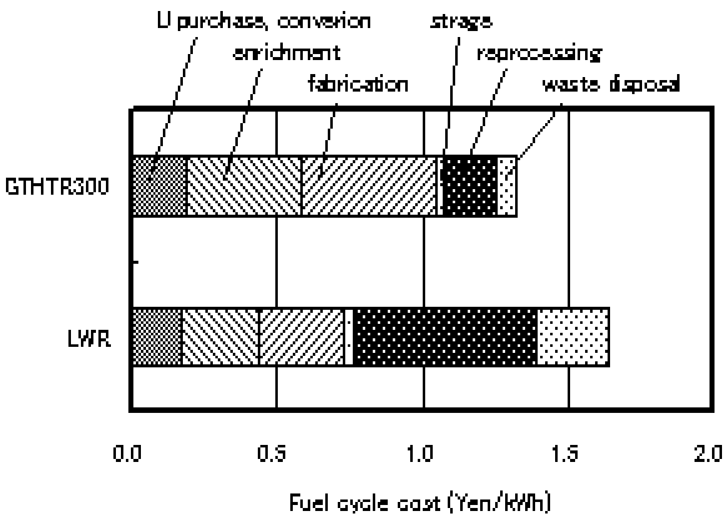

FIG. 6. Comparison of fuel cycle cost.

7. CONCLUSION

The necessary process for supplying the HTGR fuel into reprocessing plant for LWR fuels was investigated and the design of outline of the facilities was conducted. The results of this study show the technical feasibility of reprocessing of HTGR fuels by using the reprocessing plant for LWR fuels. The cost for construction and operation of these facilities was estimated and a prospect that fuel cost will be the same level or less expensive comparing with LWR fuels even if the reprocessing is considered in HTGR fuels.

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INNOVATIVE NUCLEAR REACTORS AND FUEL CYCLES FOR HIGH-SPEED CARGO VESSELS

J.A. VERGARA

Board of Directors
Chilean Nuclear Energy Commission
Santiago, Chile

Abstract. It has been about four decades since nuclear powered cargo ships were seriously discussed in the naval architecture community. Recent developments in commercial shipping include faster, bigger, and more powerful ships. In addition, several initiatives in advanced reactor design include smaller, safer and more efficient reactors. The development of advanced ships and advanced reactors opens an opportunity for nuclear propulsion. Although the goals of the INPRO and GIF Projects are not related to nuclear propulsion, these are analized for such potential case, considering the contribution to energy sustainability. These projects would help to select nuclear propulsion plant and fuel cycle concepts based on user requirements. Current technology trends point out at IPWRs and GT-MHRs, preferably the latter for overall advanced ship competitiveness. Core residual heat removal and other factors recommend the use of a hybrid power plant, with an auxiliary fossil fuel plant. This article illustrates future trends in high-performance cargo vessels, and the benefits of nuclear propulsion in such ships. Two advanced ship proposals are discussed for nuclear power, based on data availability and market opportunities. The economic viability of hybrid propulsion plants is anticipated, even with conservative reactor cost figures. The most important restriction is the reactor cost, so that these projects are opportune.

1. INTRODUCTION

Sustainable development is about the future of the overall society, regarding economic growth, environmental protection, and social equity, including access to natural resources such as food, water and energy. In fact, the Brundtland Report [1] defined it as “development that meets the needs of the present without compromising the ability of future generations to meet their own needs.” Such goal demands social and technological changes.

End-use energy consumed for heating, lighting, industry, freight and transportation will continue to expand, reaching an annual level of 18 billion tonnes of oil equivalent by the next four decades. However, about a quarter of the world has limited access to electricity, clean water and non-polluting heat [2]. The development nature of this demanding sector will finally shape the energy mix and consequent environmental degradation.

About three quarters of primary energy consumption is currently produced from the burning of limited non-renewable fossil fuels, which will continue to expand and dominate, although an internal shift from coal to natural gas is expected. Source concentration will increase fuel cargo flows to transformation centers and end-users. Transport demand, on the other hand, which relies almost entirely of oil, will be the fastest growing end-use sector, becoming the largest user in a few decades [2], so that part of liquid and gaseous fuels ought to be preserved in the mid-term.

In addition to accelerated resource depletion of energy sources that took billions of years to develop and mature, fossil fuel burning altogether will drive up carbon dioxide emissions for global warming at a rate of about 1.8% per year, doubling the current atmospheric CO_2 injection of 22 billion tonnes per year in about four decades [2]. Such environment offense can be leveled off by replacing carbon-based fuels or, in a lesser mode, by adopting end-of-pipe complex remedies such as CO_2 sequestration [3]. An apparent relief will be a progressive transition to the hydrogen economy, which only makes sense if such energy carrier is transformed from an abundant resource using a CO_2-free energy source. Sulphur dioxide and nitrogen oxides emissions, leading to acid rain, would also be reduced.

Renewable hydropower will remain comparatively small and steady but shrinking its base share in the long run. Non-hydro renewables, as a group, are required to increase sharply from a very limited current use, without taking more than a small fraction of the mid term energy mix. Energy conservation measures are to be emphasized only for the sake of avoiding any worsening energy scenario.

Nuclear energy only serves 6% of the energy mix. Expanding current nuclear technology can provide large and lasting CO_2-free energy with less harm to the environment and perhaps most importantly, with a minimum depletion of natural energy resources. Contradictorily, its potential contribution to sustainable development is unlikely without a major technological shift, resulting in much less capital intensive power plants and reduced energy production costs for competitive operation, in addition to huge reactor and fuel cycle safety, waste and non-proliferation improvements, which represent a more social oriented change in addition to that of technology.

Two of the main objectives of the INPRO project [4] are to help to ensure that nuclear energy is available to contribute in fulfilling energy needs in the 21st century in a sustainable way and to bring together both technology holders and technology users to consider jointly the international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles. The above is based on existing global energy use scenarios, which indicates that energy demand would at least double over the next 50 years. If INPRO objetives are met, a large shift from fossil fuels may become possible. In fact, innovative nuclear power plants could provide electricity for current applications in addition to a large fraction of the future transportation sector, both applied directly to batteries or via fuel cells, in addition to biomass or solar hydrogen production methods [5]. Hydrogen is also thought as aircraft fuel, subject to innovative storage technologies. Electrolytic or promising thermochemical hydrogen processes would also save natural gas and oil, for direct uses.

Hydrogen is also technically viable for future marine propulsion, which is set to increase owing to growing international trade trends, specially for non-intensive fuel consuming ships (i.e. small general cargo ships) pending storage solutions similar to those of aircraft. However, on-board innovative nuclear propulsion technologies would be better justified in some cases. Such applications demand rather low capital intensive power units, in addition to safety, waste and non-proliferation considerations.

2. NUCLEAR MARINE PROPULSION

Nuclear propulsion was born prior to nuclear power, in 1955, in the SSN "Nautilus", using an enriched uranium-fueled, pressurized water reactor (PWR), to exploit the particular nuclear marine power advantage of containing an enduring energy source in a reduced volume. Nuclear propulsion became utilized in navies of six countries (USA, UK, Russia, France, China and India) for air independent stealth in submarines and for high-speed and maximum jet fuel capacity in large aircraft carriers. A few cruisers were designed with nuclear plants due to conceptual similarity to former steam propulsion plants; however, simplicity and power density jumps of modern gas turbines, along with very stringent and expensive design requirements, progressively overshadowed their benefits in such ship types. Current naval reactors power millions of miles with a single nuclear fuel load. Moreover, a few suppliers are likely to assure a single fuel load for lifetime in naval reactors.

Commercial marine nuclear power experience did not have the naval result [6]. Stealth is not mandatory, although the so-called "green stealth" may become so in the future. Two general cargo ships (Savannah and Otto Hahn) and one experimental ship (Mutsu) were operated briefly in the sixties, mostly as nuclear technology demonstrators. Nuclear propulsion has also been used in several Russian icebreakers, in which persistance in the ice sheet has been imperative to allow cargo into the Murmansk. In fact a new unit is due to completion in 2004. Competitive nuclear marine propulsion is an option worth to be considered, provided that the reactors are designed and exploited mostly as land-based commercial nuclear utilities.

3. USER REQUIREMENTS FOR NUCLEAR MARINE PROPULSION

So far, INPRO Project has focused on the definition of prospects and potentials of nuclear power, and in the definition of user requirements for Economics, Environment, Fuel Cycle and Waste, Nuclear Reactor and Fuel Cycle Safety, Non-Proliferation and a set of Crosscutting Issues (social, infrastructure, legal, knowledge, etc.) [4]. In addition, a suitable methodology has been proposed to judge the potential of competing innovative nuclear reactors and fuel cycles technologies, over the next 50 or more years. In a similar way, Generation IV International Forum (GIF) Project [7] has gathered nuclear technology suppliers academy to set challenging technology goals in the areas of energy sustainability, economic competitiveness, safety, reliability, and proliferation resistance and physical protection, over the next 30 years. GIF has selected six system sets from almost 40 reactor and fuel cycle concepts for further research and mid term development. Although both projects have focused on land-based nuclear systems, the requirements and goals of a moving –or portable- reactor are well interpreted in their work, if such reactors are designed as land-based utilities. Most differences would rely on the characteristics and purposes of such "moving site", the power plant output and the limited plant footprint.

Economics: there are equivalent goals and requirements, perhaps differing in their product accounting (i.e., dollars per cargo mass flow transported per trip or time –$/TEU or $/tonnes– rather than dollars per energy unit produced –$/MW-h). In both uses, capital intensity must be reduced for competitiveness. INPRO requirements have focused learning rates as a measure for economic improvement. In fact, a higher learning rate for shorter follow-on plant cost reduction would certainly be better met if both land and marine technology developments are added. Both land and marine nuclear applications will be enhanced by the production of innovative modular units.

Environment, Fuel Cycle and Waste: the environmental performance requirements of a marine nuclear plant are similar to land-based reactors in a broad sense. However, particular differences are related to the fuel cycle facilities, which must be almost fully in land, including spent fuel cooling pools, so some material flows would be required. Compared to fossil power plants, there are benefits in environmental harm and resource depletion. Furthermore, at sea, CO_2 sequestration is not an alternative.

Nuclear Reactor and Fuel Cycle Safety: Reactor safety is a stringent requirement everywhere within the nuclear community in order to ensure that the general public and plant operators would not be exposed to contamination or high radiation levels. Regulatory offices are required to determine the level of safety. In particular, an international body is to be emphasized for marine applications. Added accident categories must be considered to gain acceptance, including failure or water ingress due to collision, grounding and sinking. Such are potential events for any vessel, although nowadays they are almost fully avoidable. In any case, the reactors must shut down safely in any circumstance and the probability of radioactive release should be minimized. The ship structural design and plant layout must reduce the probability of damaging the reactors. In case of sinking, the reactor would have to balance pressure in order to avoid a destructive collapse. Fuel Cycle safety requirements are similar to land-based reactors in a broad sense

Non-Proliferation: Proliferation resistance features and physical measures similar to land-based reactors can be implemented in the design, construction and operation of marine propulsion systems, assuring barrier to fissile material diversion. Particular distinctions are the intermittent absence from inspections in-route and the very low feasibility of material diversion at sea. Nevertheless, in addition to intrinsic and extrinsic measures, current reactors and marine routes can be monitored on-line at all hours by reference national or international bodies.

Crosscutting Issues: Licenses for design and operation of marine power plants must follow procedures, similar to those of land-based reactors. However, those plants should be regulated by international or multinational authorities and inspection bodies rather than national bodies. A particular condition is that the risk attributed to marine nuclear power can be handled similarly to other marine risks. Industrial infrastructure for maintenance, manning, support and research must be comparable to that of land-based reactors. Human resources and knowledge must be preserved allowing developers to prompltly recognize specific user needs and reactor or ship technology disruptions with enhanced international cooperation and hopefully common R&D platforms. Open communication between the public and other stakeholders on the safety, waste and proliferation scenarios and energy supply choices are also required in a similar way to land-based plants.

4. SHIPS FOR NUCLEAR POWER

Nuclear propulsion is an application easy to understand for air independent stealth in true submarines and for high-speed and maximum jet fuel capacity in large aircraft carriers. However, it is not so clear in commercial applications, even more if the earlier general cargo nuclear ships did not show the anticipated or publicized economic benefits of cheap nuclear power. Commercial nuclear propulsion did not show a clear success because those ships were not properly selected for economic performance, in terms of size, speed, and routes. In addition, those general cargo ships experienced socio-political problems such as public opinion concerns and environmental rejection, operator wages, etc.

There are many cargo ship types: general cargo ships, containerships, Roll-On/Roll-Off ships, barge carrying ships, and special ships. These have different sizes, cargo capacities, destinations and route frequencies. Although several recent papers have proposed small helium reactors for almost any kind of ship, not all of them may apply. User requirements (mostly infrastructure and other crosscutting issues) would limit the size, type and number of ships suitable for nuclear propulsion.

In the last years, marine transportation has shown an evolutionary trend in the high-speed transport segment, with a marked consolidation in fast ferries, and an encouraging innovation potential in other markets [8]. This evolution has been possible due to the joint emergence of new advanced monohull and multihull designs, high performance propulsors –mostly the waterjet– and highly compact power plant packages [9]. Such evolution is comparable to the aircraft evolution of the fifties, in terms of new shapes, fuselage materials and compact turbofans.

New market conditions open several opportunities. Short door-to-door transit times (i.e., a few days) are possible at very high transport fares, for small cargo inventories of highly valued goods (Figure 1). Conventional cargo airplanes are the typical platforms for such demand. On the other hand, low rates are possible subject to very long transit times (i.e., about a month) for high flows of low value commodities, using conventional cargo ships. Currently, there is a demand for fast shipping of large-packages cargo (i.e. up to 5 tons). In addition, under current globalization and enhanced foreign trade, there is a clear potential for overseas shipping of goods, which are relatively heavy or bulky for commercial airplanes, such as computers, electronics and most house appliances. On the other hand, a broad variety of new market flows can appear, such as high-value foodstuff and perishable goods, over large distances.

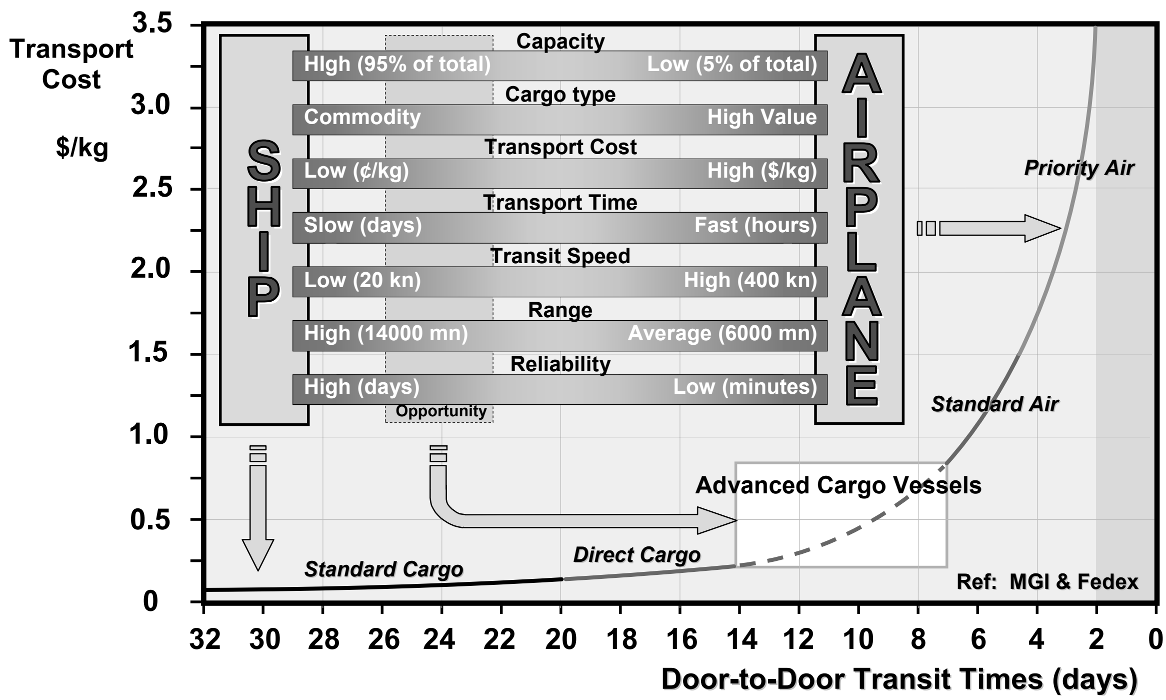


FIG. 1. High-performance cargo vessels transport cost vs transit time.

There is an opportunity gap in between, which can be covered by advanced vehicles, provided they offer high cargo capacity and intermediate speeds (i.e. about 40 knots), high reliability and moderate to high cargo value. Cargo ship types, candidate for nuclear propulsion, can be described as high-performance cargo vessels, i.e. those ships of a high load factor and cargo turnaround to render good economic results in a hypercompetitive market. Furthermore, their size (for capacity and reliability) and speed (for suitable cargo value) would be such to require a high power propulsion plant.

A powerful tool for characterizing advanced cargo platforms is the Suspension Pyramid as shown in Figure 2. The different corners of such pyramid represent pure physical means for supporting a vehicle and points inside represent combinations of those means. The most suitable ships for nuclear propulsion are those with a primary hydrostatic (Archimedean) lift with added power-based lift (hydrodynamic and/or aerodynamic). Hydrostatic lift allows higher cargo capacity but limits ship speed. In contrast, other supporting means provide reduced drag for speed at the expense of cargo and propulsion plant capacity.

The Transport Factor (TF) is another tool for characterizing nuclear cargo vessels. TF is given by the ship displacement (W) multiplied by design speed (V) and divided by shaft and lift power (P_S). Different vehicle attributes and performance can be better explained by expanding the displacement (weight) component into structures and machinery weight (W_S), cargo weight (W_C) and fuel weight (W_F) [2]. Therefore, the transport factor can be divided into ship factor, cargo factor and fuel factor.

$$TF = \frac{W \cdot V}{P_S} = \frac{(W_S + W_C + W_F) \cdot V}{P_S} = \frac{W_S \cdot V}{P_S} + \frac{W_C \cdot V}{P_S} + \frac{W_F \cdot V}{P_S} = TF_S + TF_C + TF_F$$

The former expression can be further rearranged to better identify vehicle design factors and user requirements, to render the following formula:

$$TF = \frac{W_C}{P_S} \cdot \left(1 + \frac{W_S}{W_C}\right) \cdot V + R \cdot SFC$$

The vessel design factors are systems and structures weight (W_S), nominal ship power (P_S), and specific fuel consumption (SFC), while the user requirements are cargo weight (W_C), range (R) and speed (V). The most sensitive design factors are traded-off. In fact, the higher the ship power the smaller the platform weight allowed for a given cargo. Specific fuel consumption is rather constant in fossil fuel plants for cargo vessels (at best 170 g/kW-h for medium speed Diesels and 200 g/kW-h for modern gas turbines), so that typical fuel factors depend mostly on the range. Hull shape, machinery weight and propulsor efficiency are critical in ship cargo performance. Table I shows Transport Factors for selected ship types. Notice, for example, that an eventual nuclear FastShip would have a Ship Factor of 15.5, reflecting a heavier power plant, but a Cargo Factor of 10.4, for enhanced cargo capacity and revenues.

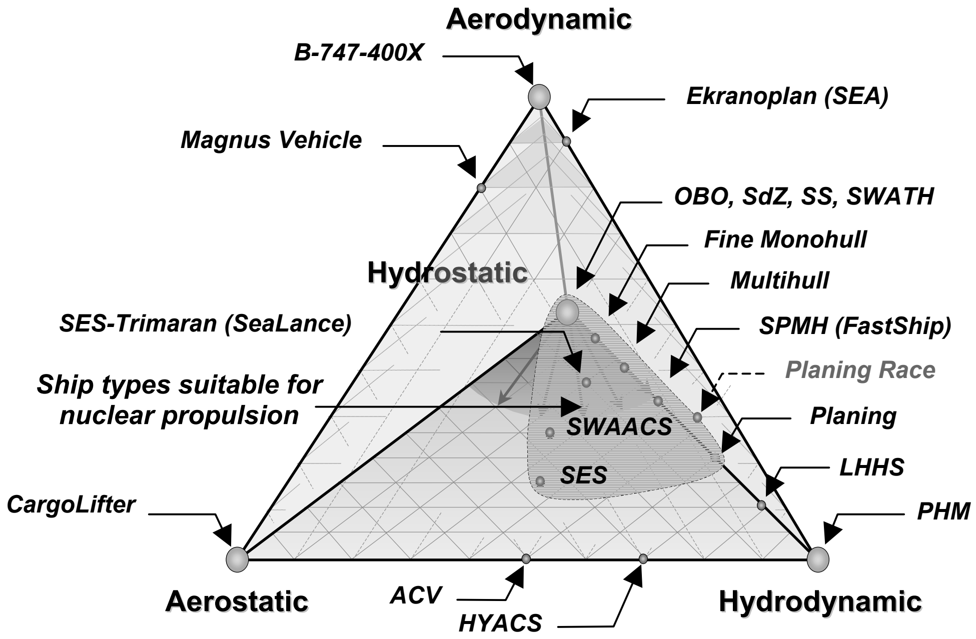


FIG. 2. The suspension pyramid (for nuclear propulsion suitability) [10].

A Volumetric Transport Factor (TF_V) can also be defined, in which internal volume is the relevant figure, instead of weight. Generally, weight is more limiting than volume in surface ships. However, a cargo submarine, i.e. for subartic cargo transport may be limited by volume rather than by weight.

Table I. Transport factors for selected ship types

Parameter \ Vehicle		SeaLance DK Group	FastShip Thornycroft	Taurus Fincantieri	Catalonia InCat	Containership Norasia	SubArtico Mitsubishi	Cargo Lifter C. Lifter AG	B-747-400 Boeing
Hull type		SES-trimaran	SPMH	V Monohull	Catamaran	U Monohull	Sumersible	Aerostat	Airplane
Length	m	200	262	136	91	174	180	260	81
Max Beam	m	32 (57)	40	22	26	28.4	25	65	6.5 (70)
Draft	m	10.5	10	4	3.6	11.2	22	-	-
Lightweight	ton	8163	15750	2700	900	9800	15000	330	125
Cargo Capacity	ton	10000	10000	1200	440	23700	26000	160	100
Fuel Capacity	ton	2828	5000	120	60	3000	0	65	200
Deadweight	ton	12828	15000	1320	500	26700	26000	225	300
Displacement	ton	20991	30750	4020	1400	36500	41000	555	470
Power	MW	155	250	70	28	9.5	40	9.3	100
Speed	kn	45	42	40	40	17	22	60	480
Machinery	-	Gas Turbine	Gas Turbine	D4G-HD	Diesel	Diesel	Nuclear	Diesel	Gas Turbine
Lift	-	32 MW (aero)	semiplaning	hydrostatic	hydrostatic	hydrostatic	hydrostatic	aerostatic	aerodynamic
Engines	-	5 x 25 MW	5 x 50 MW	2 x 22 MW	4 x 7 MW	1 x 18 MW	2 x 20 MW	2 x 5.7 MW	4 x 325 kN
Propulsors	-	5 WJ	5 WJ	4 WJ	4 WJ	1 FPP	2 FPP	2 Props	4 Fans
Range	nm	3240	3230	340	1100	9500	600000	5400	8800
Ship Factor	( )	**11.9**	**13.3**	**7.7**	**6.4**	**87.9**	**41.4**	**10.7**	**3.0**
Cargo Factor	( )	**14.6**	**8.4**	**3.4**	**3.1**	**212.6**	**71.7**	**6.2**	**2.4**
Fuel Factor	( )	**4.1**	**4.2**	**0.3**	**0.4**	**26.9**	**0.0**	**2.1**	**4.8**
Transport Factor	( )	**30.5**	**25.9**	**11.5**	**9.9**	**327.4**	**113.0**	**17.9**	**11.3**

The fuel weight is irrelevant in nuclear propulsion, regardless the users range requirement, since new fuel is loaded as a single batch every year and a half or even less often, depending on nuclear fuel design and burn-up strategy. Such low fuel consumption is an important feature for efficient cargo transport. It also allows very long ranges. In some cases, the fuel factor may be very large (and indeed very expensive) and it may be burned with such rate to limit endurance, so that nuclear propulsion becomes a clear advantage, allowing high revenues. The overall advantage is the net balance between fossil fuel weight savings and weight additions for the reactor, shielding and ship structure reinforcement, which may compromise the ship weight factor. Figure 3 shows the relationship between TF and speed for different vehicles.

In terms of economics, nuclear propulsion is a viable alternative where very high fossil fuel consumption is expected, since the nuclear advantage relies on the lower cost of the fuel component of the overall ship operation cost. However, such advantage stands for high service factor, not encouraging coastal trade. In addition, specific nuclear plant costs typically double those of gas turbines, so that maximum plant cost optimization is mandatory. Modular commercial plants should be utilized for low capital costs.

However, there are other factors to consider. For example, infrastructure user requirements may limit the number of ports with maintenance and general repair capabilities for nuclear cargo ships, and for nuclear refueling or fuel repairing. Besides, large ships are less sensitive to bad weather, so that reactor safety risks may be somewhat lesser in larger ships. Non-proliferation concerns may become a limiting political factor for defining ship type and flag and the number of units worldwide, regardless the strength of the protection measures. Public opinion may also restrict the number of countries and ports open for nuclear powered cargo ships, so that a concept of port hubs is more suitable for nuclear ship cargo systems. In addition, ship operation costs are sensitive to human resources and qualifications, so that operators should expect higher manning costs –because of higher wages– with nuclear power plants than with other propulsion means. The ship should be structurally reinforced and equiped to minimize watern ingress in case of collision or groundiung the the reactor room.

Although several papers have proposed very small reactors suitable for almost any kind of ships, it is expected that the above user requirements and economic factors would limit the type, size and number of working nuclear powered cargo ships. Small and/or slow ships demand low power, so that these may be better arranged with conventional medium speed engines, burning fuel oil or natural gas, preserved or saved by nuclear power production. In an utmost constrained environment, low power intensity users (i.e. cars, trucks, trains, small ships, etc.) should consume hydrogen, using fuel cells of modified turbines, or other energy carriers (i.e. compressed air, inertia wheels, etc.). For optimum environmental conservation, such energy carriers should be transformed or produced from nuclear facilities and renewable energy.

Therefore, nuclear propulsion is best suited for high-performance cargo vessels, i.e. those ships of high cargo turnaround, load factor and range, with such a ship size or speed to demand a very high power propulsion plant, i.e. one with a power demand in excess of 100 MWe. For reference, the most powerful compact gas turbines available deliver about 50 MWe each one (with a footprint of ~40 m^2, and a specific weight of 2-5 kW/kg), and the most powerful medium speed Diesels available deliver about 35 MWe each one (with a footprint of ~110 m^2, and a specific weight of 0.05-0.2 kW/kg). Fossil fuel consumption for 100 MWe would be about 500 tonnes per day, a large fraction of the cargo deadweight.

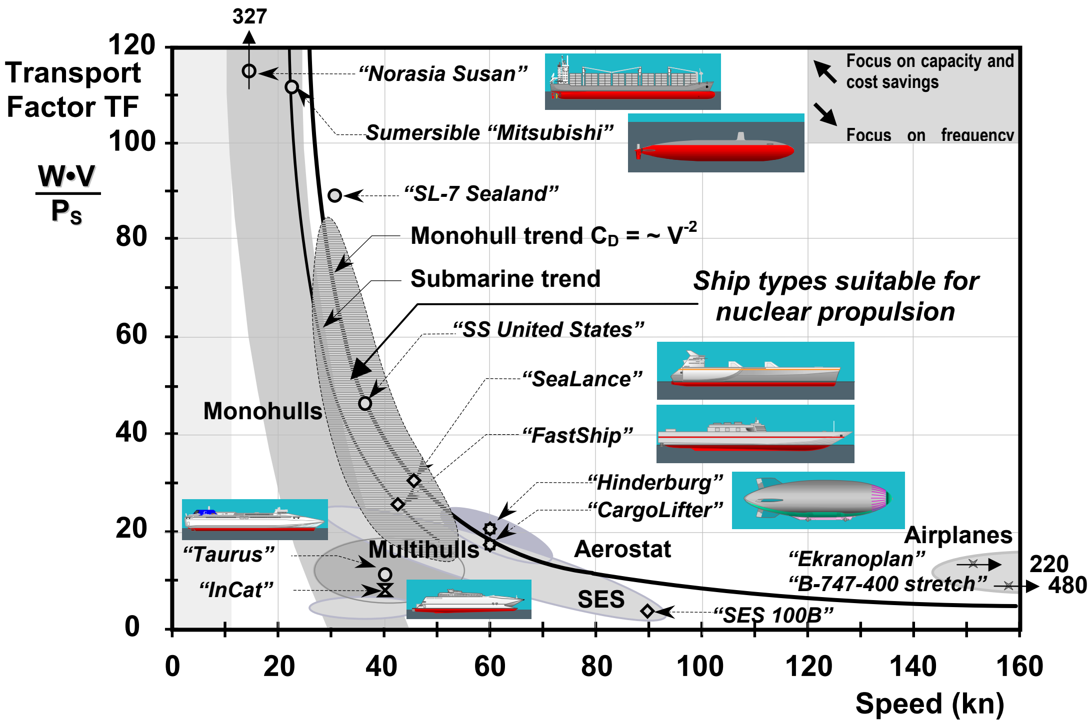


FIG. 3. Transport factors for nuclear powered transoceanic cargo vessels [10].

5. REACTORS AND FUEL CYCLES FOR MARINE POWER

Almost all naval and marine nuclear propulsion plants have been conceptually similar to that of the SSN "Nautilus" (with PWRs) with technology evolution at the component level (fuel endurance, pump noise, shock absorbers, steam generator layout, etc.). Such reactors have intrinsic limitations for compactness and efficiency, mainly the excessive vessel weight required for an efficient steam cycle in single-phase flow and core shielding. An exception has been the liquid metal cooled reactor (LMR) that takes advantage of the large heat transfer capacity of liquid metals, i.e. lead-bismuth eutectic, and the vapour pressure of the coolant, enabling compact reactors. LMRs have been used in high-performance Russian submarines (i.e. Alpha class), and was proven firstly in the second nuclear submarine (SSN "Seawolf") of history, in the mid fifties, but it was earlier converted to PWR due to concerns of chemical reactivity of the sodium coolant, with water ingress [11]. Not many of such Russian submarines are still in service.

Unlike gas turbine and Diesel engine development, nuclear technology has been conservative to date. Newer reactors are mostly evolutions of proven designs, despite the broad innovation potential in the field. Reactor design has evolved from small power, suitable for nuclear propulsion, in the early sixties to very large size (i.e. beyond 1500 MWe) searching for size-driven economies of scale, leaving nuclear propulsion only the conservative naval sector. Such approach is technically correct; however, the learning curve effect should be better based on the delivery of smaller scale reactor units, which are integrated in a single station for similar energy production or power output. Maintenance and repair can be better managed using modular units. Station load factor can also be better scheduled.

Furthermore, large scale has resulted in the use of very expensive redundant safety systems, offsetting potential economic savings. In addition, large and complex power plant construction took much longer, increasing owners´ risk and loan interest, rising capital and operation costs, and started to evaporate the promise of cheap nuclear energy. Such was the status at the outset of the oil crisis, which was further affected by the introduction of inexpensive combined cycle gas turbines, not mentioning TMI and Chernobyl accidents, the expansion of the anti-nuclear sequel, and other political decisions regarding nuclear waste isolation, fuel cycle, reprocessing, etc..

The current state of nuclear technology in a fossil-driven energy system, has restated the idea of medium and small nuclear reactors (SMR) [12], and at about the same time idea of modular plants, in order to reverse the cost spiral and obstacles. An interesting fraction of the capacity under construction is based on SMRs. Recent INPRO and GIF Projects have shortened the gap to real innovative modular reactors of simpler design, mass production economics, reduced siting costs, long life cores, unattended operation and enhanced safety performance. INPRO has defined prospects, potential and user requirements for long-term nuclear power generation, process heat, district heating, desalination and waste transmutation with associated fuel cycles while GIF has defined more specific systems for the next few decades.

So far, none of them have inspired propulsion reactors. Competitive nuclear marine propulsion is an option worth considering for ships of the type mentioned above, with smaller power than commercial reactors and high service factors, provided that such reactors are designed and exploited mostly as land-based commercial nuclear utilities, for learning curve economics. Special features are the design and operation of a moving reactor, safety measures to prevent grounding, reactor pressure balance in case of sinking, and means to safely transfer spent fuel to land based fuel cycle facilities, in a limited footprint. Figure 4 sintetizes Reactor and Fuel Cycle families selected by the INPRO and GIF Projects, downsizing from several tens of reactor-fuel cycle types, viewed from both suppliers and users sides. Nuclear fuel cycles demand further R&D for non-proliferation, sustainability and environment [13].

The dominance of PWRs in nuclear power generation and propulsion is a resilient consequence of the early selection of water as a coolant, due to its moderation properties, pumping versatility, heat transfer capacity, chemical stability, availability and cost. PWRs are robust and proven reactors for aircraft carrier and submarine propulsion; however, specific volume and weight may seem limiting for merchant ship propulsion. Integrated PWRs, radically simplified, may satisfy marine propulsion user requirements. Units of that sort are installed in recent aircraft carriers and submarines, although their specific economics and weight may not be suitable for high-performance ships. Cargo ships may incorporate IPWRs such as CAREM-F, IRIS, MRX (all three up to 300 MWe) or SMART (100 MWe) [14-16]. In terms of desired features, IPWRs should include in-vessel steam generators, pressurizer and CRDM, and compact water filled containment. The best of such reactors would have a specific weight of 0.03 kW/kg.

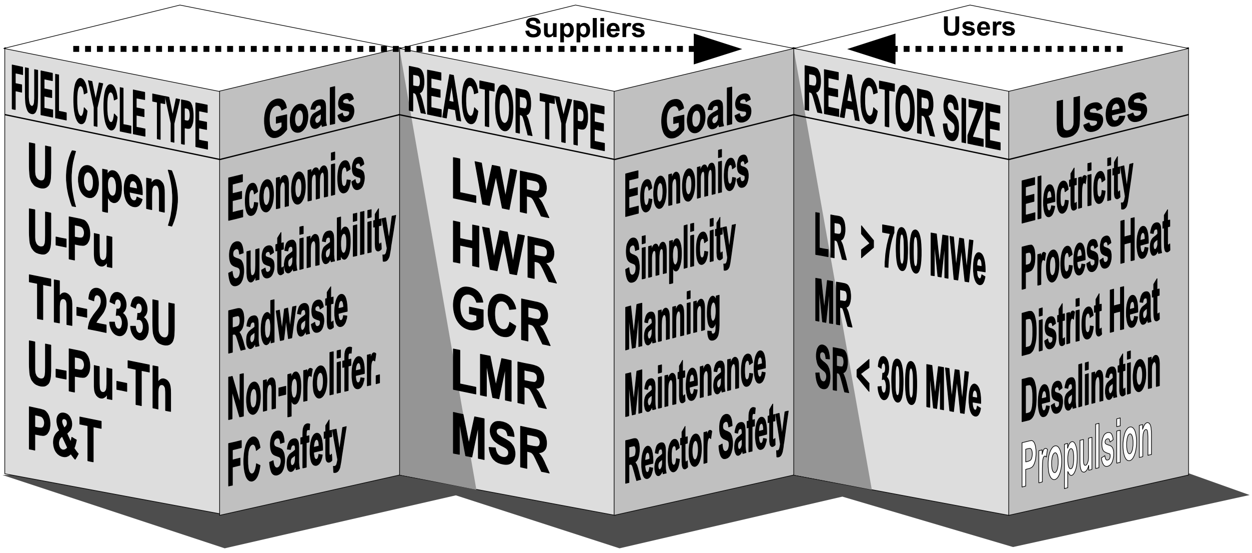


FIG. 4. Nuclear reactor and fuel cycle types, goals, sizes and applications.

Another reactor choice for commercial marine applications is the GT-MHR (Gas Turbine Modular Helium Reactor) proposed by General Atomics, a derivative of the gas-cooled reactor (GCR). Such reactor system offers high efficiency, chemical fuel-to-coolant affinity, safety [17] and fuel burnup [18], resulting in a light, inexpensive and simple plant [19]. It combines the experience of the high temperature GCR and compact heat exchangers, with that of aeroderivative and industrial gas turbines. There is another GCR, the PBMR (Pebble Bed Modular Reactor) designed by Eskom, which might lead the commercial reactor version. Both concepts are arranged in a closed Brayton thermal cycle for higher thermal efficiency (47%), but differ basically in the nuclear fuel system. The PBMR has a pebble bed core, consisting of a stack of thousands of 6 cm diameter graphite balls with TRISO uranium micro-particles within them. Suitable for power generation, it can be refueled on-line with a recharging and monitoring system which could even provide a reflected annular core configuration [20]. However, such system puts an additional complexity to propulsion power plant operation, and to international safeguards.

The main attribute of a GT-MHR, provided that it has a low power density, is its capacity to tolerate a full loss of coolant without core meltdown (a critical factor in reactor licensing and public opinion) and the stability of the coolant. Safety resides in a microencapsulated fuel that can retain fission products during such an accident, its capacity to passively shut down the reactor if temperature increases, and a safety-related favorable core geometry. Intead of being used in graphite balls, microparticles are encapsulated in compact rods and arranged within hexagonal-prismatic graphite blocks [21]. A marine GT-MHR would be less sensitive to ship motion than a IPWR, lighter (0.09 kW/kg), while its power conversion units would provide direct torque as helium expands in the turbines, in addition to cost advantages [22-23]. Such reactor may be one of the most convenient reactors for high-performance cargo vessels.

Other reactor systems do not show as promising for propulsion of such ships as IPWR and GT-MHR. For example, boiling water reactors (BWR) were supposed to offer slight investment gains, but it has not found justification in propulsion, because of coolant phase changes and heaving accelerations that would impair reactor power control, plus extra shielding due to radiochemical carryover. Supercritical reactors (SCR) do not experience phase changes and allow high thermal efficiency. However, materials for severe corrosion degradation of very thick vessels are not yet fulfilled. Heavy water reactors (HWR) are large and use complex on-line refueling systems. Molten salt reactors (MSR) are good for safety, waste performance and resource sustainability. However, plant operation and fuel treatment at sea may be too complex. Advanced LMRs are fine for sustainability, non-proliferation and co-generation. However, neutron spectrum, fuel design and other issues makes them less desirable for high-performance ships.

An inconvenience of nuclear power propulsion, especially IPWRs, is the necessity of core residual heat removal. Even after reactor shutdown, the core remains hot and requires a cooling flow, with several auxiliaries in service. Such restriction may recommend the use of an auxiliary fossil powered plant for port manouvering, emergency reactor cooling and power sprint. Hybrid power plants, including a number of gas turbines, would satisfy the above restrictions. In addition, these cargo ships would not be required to operate their reactors at port resulting in smoothing down public opinion.

Several combinations of prime movers, propulsors and power conversion mechanisms are possible. A power conversion vessel (PCV) is a device containing a set of turbines, compressors and heat exchangers arranged to obtain the best thermal efficiency. Power can be drawn directly through a helium power turbine, or electrically through a vertically mounted generator-PCV set.

6. POTENTIAL HIGH-PERFORMANCE CARGO VESSELS

There are several remarkable ship proposals in the containership segment, using architectures shown in Figures 2 and 3. Some of those ships remain controversial while designers evaluate and discuss their performance, attracting the attention of shipping business actors and media.

An interesting proposal has been the FastShip concept shown in Figure 5, a large size semiplaning monohull for high-speed transatlantic cargo. This ship would require five gas turbines driving 50 MWs waterjets for sailing at about 40 knots [24]. The author had independently proposed a nuclear version for such concept and found economic and environmental benefits, related to data published elsewhere [25].

FIG. 5. Monohull and multihull high-performance cargo ships.

Another remarkable proposal in the containership segment has been the Pentamaran concept developed by Nigel Gee and Associates, a large size multihull for high-speed transatlantic cargo. In fact, it is a trimaran with two additional outrigger sponsons "in stand-by" for stability. The center hull of a multihull is typically very slender for reduced hydrodynamic resistance. As a comparison, the beam of the FastShip would be about 40 meters. Although there is not much data available for the Pentamaran in the open literature, it is possible to argue that it would require about 100 MWs for about the same cargo capacity and speed of FastShip. That power demand could be satisfied with two large gas turbines driving 50 MWs waterjets. The author has not estimated the economic figures for the Pentamaran. However, it is expected that this vessel would also have economic and environmental benefits using nuclear power, although based on the relative power consumption, these will not have the observed gain of a nuclear monohull.

A drawback of nuclear propulsion is the reactor cost. Assuming a very conservative specific cost of $2,800/kWs (almost three times the reactor cost target set at the INPRO and GIF Projects), including reactors, containments and auxiliaries, would render a nuclear plant cost of about $700 millions for the monohull. A hybrid plant for that ship would cost sensibly less. The advantage is in the operation cost. All cost components considered, the operation cost of a nuclear or hybrid monohull would be at least 15% lower than that of the fossil counterpart. Furthermore, since uranium cost is about 5% of the power generation cost, marine cargo costs would be very stable to any sharp energy resource fluctuation. Fuel oil cost is unstable during crisis and a considerable small variation is enough to render red numbers when the fuel cost component is about 30%, a typical figure in fast cargo [25].

Figure 6 shows alternative power plants for ships similar to those concepts. For example, a thirty thousand tonnes fast monohull could be driven by a set of four 50 MWs helium gas turbines directly coupled to two MHRs, each one driving a set of two PCVs. A ten thousand tonnes monohull could be driven by a set of two helium gas turbines directly coupled to one MHR driving a set of two PCVs. In each case, a gas-oil gas turbine is placed at the centerline for port manouvering and back-up. Conversely, a multihull could not arrange all propulsors in the center hull, so that the outer ones could be arranged at the main outriggers. For such purpose, power conversion should be electric, so that a thirty thousand tonnes fast multihull could be driven by a set of two electric motors electrically coupled to one GT-MHR driving a vertically mounted generator. A seventy thousand tonnes multihull could be driven by a set of four electric motors electrically coupled to two GT-MHRs, each one driving a generator. Similarly, a fuel-oil gas turbine is considered in each case for port manouvering and back-up.

7. RELATIVE ENVIRONMENTAL IMPACT

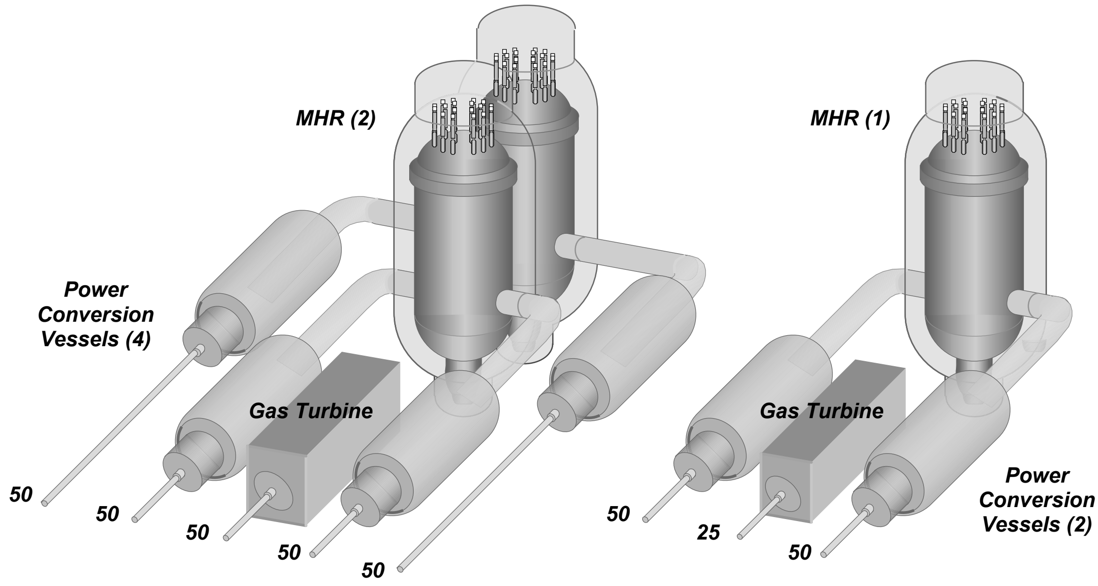
Nuclear reactors are environmentally friendly power systems with respect to fossil-fueled plants, since these units do not release combustion products to the atmosphere, an advantage that might revitalize nuclear power and nuclear propulsion if global warming proves to be as severe as anticipated. As shown in Table II, a 250 MWs monohull with five gas turbines, would emit about one million tons of CO_2 and fifteen thousand tons of other gases per year.

The environmental impact of a nuclear or a hybrid monohull would be a few tons of solid spent fuel per year, of which most could be recycled, leaving less than a ton per ship-year confined for long-term decay. The environmental response of GT-MHR is expected to be better than that of LWRs due to enhanced uranium utilization, higher burn-up, better uranium confinement, potential direct nuclear waste disposal forms and a few other advantages, which have not yet been confirmed.

Table II. Relative Environmental Impact (yearly basis) for a Fast Monohull [25]

Factor/Emission	Gas Turbine	Hybrid	Nuclear
Fuel feed (Tons)	350,100	33,900	
Nuclear feed (Tons)		5.0	5.6
SO_2 (Tons)	6,700	645	
CO_2 (Tons)	1,000,200	96,700	
NO_X (Tons)	8,300	806	
Particulate (Tons)	500	48	
Nuclear HLW (Tons)		4.7	5.3
Persistent nuclear waste		0.18	0.21
Thermal efficiency	41%	46.9%	47.6%
Heat waste (MW)	323	256	247
Noise	high	Intermediate	low

Nuclear propulsion would be coherent with the objectives of the INPRO and GIF Projects. In addition to deferring global warming, nuclear propulsion would save about five thousand tons of oil per trip (about 350 thousand tonnes per ship-year) for direct uses, for a 250 MWs monohull.

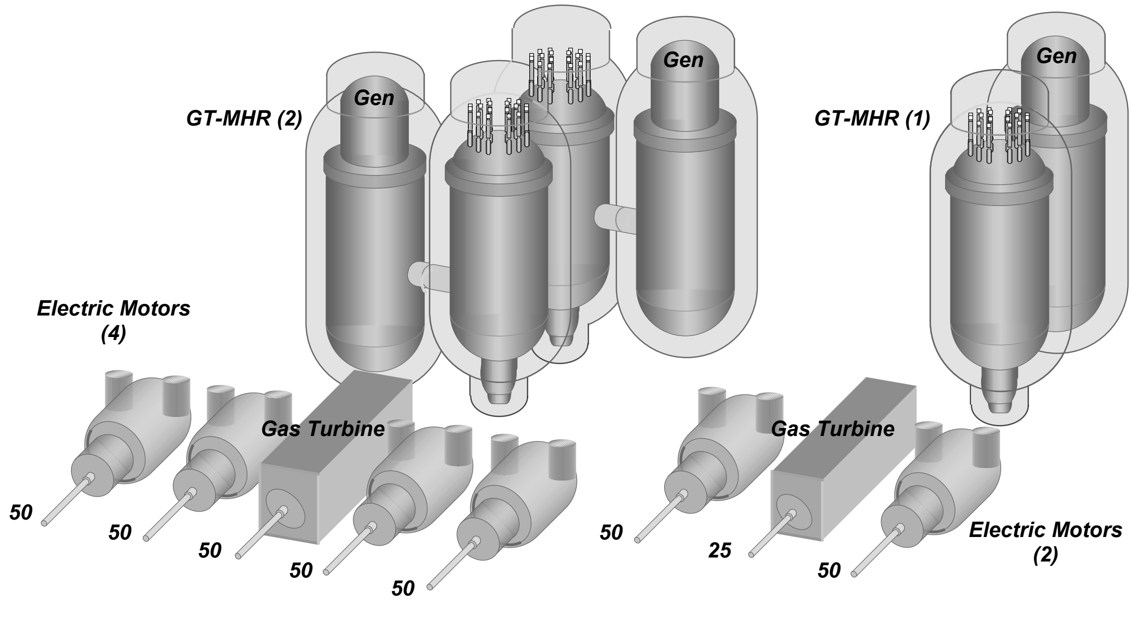


200 MWs cruise (250 MWs maximum)

High Power Demand Monohull

100 MWs cruise (125 MWs maximum)

Medium Power Demand Monohull



200 MWe cruise (250 MWe maximum)

High Power Demand Multihull

100 MWe cruise (125 MWe maximum)

Medium Power Demand Multihull

FIG. 6. Hybrid reactor configurations for monohull and multihull ships.

8. CONCLUSIONS

Two of the main objectives of the INPRO Project are to help to ensure that nuclear energy is available to contribute in fulfilling energy needs in the 21st century in a sustainable way and to bring together both technology holders and technology users to achieve innovations in nuclear reactors and fuel cycles. On the other hand, GIF has selected six reactor and fuel cycle concepts, based on sustainability, economics, safety and non-proliferation. If INPRO and GIF objetives are met, innovative nuclear power plants could provide massive direct and indirect electricity for mankind consumption.

The current state of nuclear technology in a fossil-driven energy system, has restated the idea of SMRs and modular plants. Recent INPRO and GIF Projects have shortened the gap to real innovative modular reactors of simpler design, mass production economics, reduced siting costs, long life cores, unattended operation and enhanced safety performance. INPRO nor GIF have inspired propulsion reactors. However, land-based nuclear systems requirements and goals can be "adapted" to support propulsion.

This article was intended to illustrate the future trends in high-performance cargo vessels, and the benefits of nuclear power in propulsion of such ships. Nuclear propulsion is best suited for those ships of high cargo turnaround, load factor and range, with such a ship size or speed to demand a very high power propulsion plant, i.e. one with a power demand in excess of 100 MWe. Suitable ships are those with a primary hydrostatic lift with added power-based lift, for maximum cargo transport factor.

Integrated PWRs may meet marine propulsion user requirements. Certain large size cargo ships may incorporate reactors such as CAREM-F, IRIS, or MRX, and most high-performance cargo vessels may accommodate reactors similar to the less expensive GT-MHR, for high thermal efficiency, fuel-coolant affinity, safety, and fuel burnup. An inconvenience of nuclear propulsion is the core residual heat removal. Such restriction, and potential public oppinion concerns, recommends the use of a hybrid power plant, with an auxiliary fossil fuel plant.

A marine GT-MHR concept, with several options for power conversion, would be lighter and less sensitive to ship motion than a IPWR. This nuclear plant would be almost as heavy as a Diesel power plant and several times heavier than advanced gas turbine plants. The plant volume is comparable to that of gas turbine plants if exhaust gas and intake air ducts are considered. However, such ships may carry several times their propulsion plant weight just in fuel. Nuclear propulsion would save part of that fuel weight for exhanced cargo load.

Two advanced ship design proposals are discussed for nuclear propulsion based on data availability. The economic viability of a hybrid GT-MHR-gas turbine propulsion plant for high-performance cargo vessels is anticipated, added cargo capacity, higher load factor and relative stability to transport rates and fuel volatility, even with very conservative figures. Also, nuclear propulsion could open markets beyond the ones foreseen with gas turbines, due to increased range. The most important restriction is the reactor cost, so that INPRO and GIF Projects come at the proper time for a revaluation of nuclear propulsion for commercial ships. These projects may also influence naval nuclear propulsion plant designs.

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ANALYSIS OF FUTURE NUCLEAR POWER PLANTS COMPETITIVENESS BY USING STOCHASTIC METHODS

D. FERETIC, N. CAVLINA, Z. TOMSIC

University of Zagreb
Faculty of Electrical Engineering and Computing
Department of Power Systems
Zagreb, Croatia

Abstract. The paper describes the application of a stochastic method for comparing economic parameters of future electrical power generating systems including conventional and nuclear power plants. The input data for calculation are cost elements defined in the frame of certain limits of expected occurrence with a probability distribution within the limits. The method is applied to establish competitive specific investment costs of future nuclear power plants when compared with combined cycle gas fired units combined with wind electricity generators using best estimated and optimistic input data. The results of analysis showed that future competitive nuclear power plant specific investment cost would be not considerably different from presently expected values.

1. INTRODUCTION

The increase of electricity demand is linked with the development of economy and rise of living standard in each country. This is especially true for developing countries with electricity consumption far bellow average electricity consumption in industrialized countries.

To satisfy the increased demand it is necessary to build new electrical power plants, which could in an optimal way meet, the imposed acceptability criteria. The main criteria are potential to supply the required energy, to supply this energy with minimal (or at least acceptable) costs, to satisfy licensing requirements and be acceptable to public. The main competitors for unlimited electricity production in next few decades are fossil power plants (coal and gas) and nuclear power plants. New renewable power plants (solar, wind, biomass) are also important but due to limited energy supply potential and high costs can be only supplement to the main generating units. Large hydropower plans would be competitive under condition of existence of suitable sites for construction of such plants.

Taking the above into consideration it is obvious that planning of future electricity generation must be based on fossil and nuclear technologies. Presently among these the preference is given to gas fired combined cycle plants. The preference is considered in relation to costs, public acceptance and environmental impact (in comparison with coal fired plants).

The cost of electricity generation is certainly a condition of major importance to accept a power plant project. In order to make cost calculation fair the foreseeable costs items should be considered during plant lifetime, discounted to beginning of plant operation and divided with the discounted energy produced in the same period (discounted energy is proportional to the discounted profit).

Each cost term has a certain uncertainty margin, which has to be foreseen by using best estimate approach. The problem to be considered is how to define the uncertainty distribution of calculated electricity costs by taking into account these uncertainty margins. Using in calculation random values of each cost items within given uncertainty frame (Monte Carlo method) solves problem. Large number of calculations to obtain the result requires use of suitable computer code.

This paper summarizes the methodology and present selected results of the probabilistic analysis of the performance and costs and compare the coal fired power plant, gas fired power plant and nuclear power plant.

The purpose is to assess the uncertainty of several key performance and cost of electricity produced in coal fired power plant, gas fired power plant and nuclear power plant developing probability distribution of levelized price of electricity from different Power Plants, cumulative probability of levelized price of electricity for each technology and probability distribution of cost difference between the technologies. The key parameters evaluated include:

— levelized electrical energy cost US$/kWh
— discount rate
— interest rate for credit repayment
— rate of expected increase of fuel cost
— plant investment cost US$/kW
— fuel cost US$/GJ
— constant annual operation and maintenance cost US$/kW
— variable maintenance and operational cost (no fuel)
— load factor
— plant efficiency
— years of credit repayment
— years of plant life time.

In considering feasibility of innovative Nuclear Power projects based on advanced reactor technologies one of key issue is economical is competitiveness of such projects compared with alternative conventional power plants. From the point of view of energy availability economics, development status presently main competitors to nuclear plants are gas fired plants with combined cycle, conventional or advanced coal fired plants and power plants based on renewable energy sources (among those for electricity production presently seem that most competitive are wind power plants). The aim of this paper is to analyse the competitively of power plant projects being technically and economically feasible for application toward the end of this decades. The analysis is based on comparison of discounted lifetime costs of different alternatives. Since prediction the future costs (specially fuel costs) is highly speculative the best approach is to apply a probabilistic method of costs calculation based upon expected best estimate ranges of cost components and if feasible probability distribution within the range. Such method allows to apply a Monte Carlo approach to establish set of random input data taking into account given ranges of parameters and probability distribution within the ranges. Such input data are then used for calculating produced electrical energy cost. By repeating the calculation several thousand times is possible to obtain a distribution of energy cost probability for a specific power technology and corresponding most probable value of such cost. There are available several computer programs suitable for described analysis. In our analysis we used program STATS developed at the Argonne National Laboratory (USA).

2. PROBABILISTIC ANALYSIS METHODOLOGY

The Probabilistic Analysis is performed in three steps:

(i) The analysis determines the expected range of uncertainty for key design and economic variables that make the greatest impact on the levelised cost of electricity
(ii) Developing a probability distribution for each key input variable
(iii) Monte Carlo analysis generates a probability distribution for each key performance and cost parameter using developed probability distributions.

The Monte Carlo analysis generates the probability and cumulative probability distribution for levelized cost of electricity for each technology using the probability distribution developed for the input variables. The analysis generates a large number of random samples of the input variables and corresponding value of electricity levelized generating cost and generate the probability distribution by "counting" the number of times each value of the performance parameter occur. The results are then sorted and plotted in the form of probability distribution and cumulative probability distribution.

In STATS model an assumption for value is defined by choosing a probability distribution that describes the uncertainty of the input data. In STATS model it is possible to chose between 3 distribution types – Uniform distribution, Triangular distribution and Five points distribution. Figure 1. shows the types of relative probability density functions that can be used in STATS model.

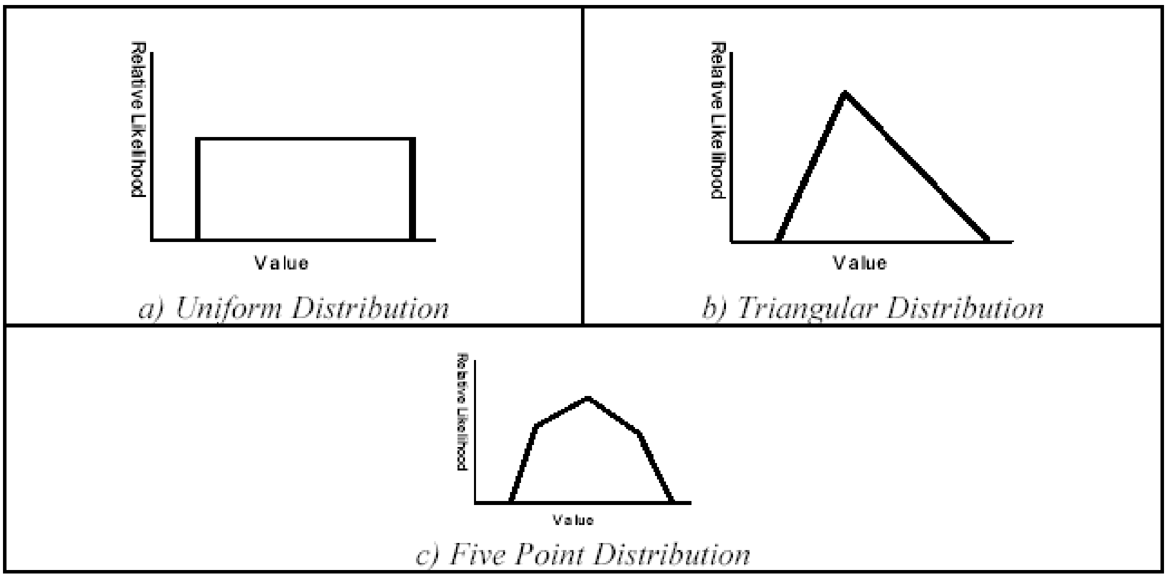

FIG. 1. Different types of relative probability density functions.

3. METHOD APPLIED IN THE ANALYSIS

The method applied in the analysis consists in comparing levelized plant life time cost for alternative technologies. These costs are usually defined for two periods. The first period is corresponding with the time of capital repayments and the second with the time of plant operation following the capital repayment. The cost of produced energy t during these period are given by the expression (1):

$$c_e = \frac{\sum_{n=1}^{n_{cr}} \frac{1}{(1+p_d)^n} \left[\frac{p_c c_i}{1-(1+p_c)^{-n}} + c_{cm} + 8760 l_f \left(\frac{c_f (1+p_f)^n}{277{,}8\eta} + c_{vm} \right) \right]}{\sum_{n=1}^{n=n_{lt}} \frac{8760 l_f}{(1+p_d)^n}} + \frac{\sum_{n=n_{cr}}^{n_{lt}} \frac{1}{(1+p_d)^n} \left[8760 l_f \left(\frac{c_f (1+p_f)^n}{277{,}8\eta} + c_{vm} \right) + c_{cm} \right]}{\sum_{n=1}^{n_{lt}} \frac{8760 l_f}{(1+p_d)^n}} \tag{1}$$

where

c_e - levelized electrical energy cost US$/kWh
p_d - discount rate
p_c - interest rate for credit repayment
p_f - rate of expected increase of fuel cost
c_i - plant investment cost US$/kW
c_f - fuel cost US$/GJ
c_{cm} - constant annual operation and maintenance cost US$/kW
c_{vm} - variable maintenance and operational cost (no fuel)
l_f - load factor
η - plant efficiency
n_{cr} - years of credit repayment
n_{lt} - years of plant life

4. INPUT DATA

Estimation of reliable input data is a condition to obtain realistic results of the calculation. The advantage of used method is the fact that the data need not to be specified as constant values but estimated within certain ranges. This allows to define a probability distribution of calculated value.

The mathematical expression given above assumes knowledge of plant operational costs and amount of produced energy for each year during plant operation. The prediction of future costs and produced energy is obviously not possible. Is possible however to estimate changes of cost elements which are of essential importance for energy costs and which do not depend upon specific plant operational conditions but are influenced by international market conditions. When considering cost of near future electricity production the key cost element is natural gas cost. In addition an intensive introduction of renewable energy sources is conditioned with reduction of their investment cost due to learning process. The most favourable renewable energy source is presently wind power. Learning coefficients, bases primarily on Danish and German experience for this energy source (learning coefficient are around 0,96), were considered in predicting their future investment costs.

4.1. Future natural gas costs

Future natural gas costs for electricity generation will with large probability have tendency to increase with larger rate than costs of other fuels. All cost predictions are based on constant dollar value. The reason for such forecast are:

- Scarcity of domestic natural gas production in West European countries and in US and large dependence upon gas imports. By an analysis presented at the recent Word Energy Conference the deficit (difference between demand and own gas production) in West Europe around 2030 could reach 300-500 billions cubic metres per year.
- Increased gas demand will cause the necessity of large investments in new gas lines connecting consumption centres with gas fields. For European consumers the main future sources of gas supply will be in Russia and Middle East. Prediction of investments needed for new gas lines in next two decades will are of order 300-350 billion dollars. These investments jointly with the fact that the gas market will become more monopolistic will certainly stimulate increase of gas cost.
- Gas consumers are not only utilities but also industries and households which are not flexible in switching fuel. These consumers have priority in gas supply because they can accept higher gas costs. This fact will limit gas supply to utilities in the case of increased demand.

For the specifies reasons the best estimate for gas cost at the end of this decade ii Europe is of the order 4 US$/GJ with further increasing rate 3-5% per year. Optimistic assumption is 3.5 US$/GJ with increasing rate 2-4%.

It is reasonable to predict that the prices of other fuels (coal, nuclear) will be considerably more stable than for natural gas because of larger reserves and smaller demand.

Input data including best estimated and optimistic costs predictions are used for comparison of electrical energy costs produced in nuclear, coal and wind and natural gas plants. These data are given in Table I and II.

Table I. Best estimated input data for evaluating distribution of electrical energy costs of nuclear and coal power plants build around year 2010

Power Plant	**Nuclear**	**Coal**
Total investment cost US$/kW		
Probability distribution	Triangular	Triangular
Limit values	1900 2000 2100	1400 1500 1600
Fixed operational and maintenance costs USD/kW year		
Probability distribution	Uniform	Uniform
Limiting values	100 120	30 40
Average interest rate		
Probability distribution	Uniform	Uniform
Limiting values	0.055 0.075	0.055 0.075
Loan repayment period (years)		
Probability distribution	Uniform	Uniform
Limiting values	15 20	15 20
Capacity factor		
Probability distribution	Triangular	Triangular
Limiting values	0.6 0.7 0.8	0.5 0.6 0.7
Fuel cost USD/GJ		
Probability distribution	5 points	5 points
Limiting values (relative probabilities)	0.45 0.47 (0.7) 0.5 (1.0) 0.52 (0.7) 0.55	1.8 1.9 (0.7) 2.0 (1.0) 2.1 (0.7) 2.2
Plant efficiency		
Probability distribution	Uniform	Uniform
Limiting values	0.32 0.34	0.38 0.42
Variable operational and maintenance costs (no fuel) US cents/kWh		
Probability distribution	Uniform	Uniform
Limiting values	0.15 0.25	0.30 0.40
Discount rate		
Probability distribution	Uniform	Uniform
Limiting values	0.05 0.08	0.05 0.08
Fuel cost increase rate		
Probability distribution	Uniform	Uniform
Limiting values	0.008 0.01	0.01 0.02

Table II. Best estimate and optimistic data for natural gas combined cycle plant and wind powered plant build around year 2010

Power Plant	**Wind**			**Natural gas-combined cycle**				
Total investment cost US$/kW								
Probability distribution	Triangular			Triangular				
Limit values (best estimate)	700	800	900	500	600	700		
Limiting values (optimistic)	600	700	800	400	500	600		
Fixed operational and maintenance costs USD/kW year								
Probability distribution	Uniform			Uniform				
Limiting values	10	15		10	20			
Average interest rate								
Probability distribution	Uniform			Uniform				
Limiting values				0.06	0.08			
Loan repayment period (years)								
Probability distribution	Uniform			Uniform				
Limiting values	12	15		12	15			
Capacity factor								
Probability distribution	Uniform			Uniform				
Limiting values	0.2	0.3		0.75	0.85			
Fuel cost USD/GJ around year 2010								
Probability distribution				5 points				
Limiting values, best estimate (relative probabilities)				4.0	4.2 (0.7)	4.5 (1.0)	4.75 (1.0)	5.0
Limiting values, optimistic (relative probabilities)				3.0	3.2 (0.7)	3.5 (1.0)	3.75 (0.7)	4.0
Plant efficiency								
Probability distribution				Uniform				
Limiting values				0.54	0.62			
Variable operational and maintenance costs (no fuel) US cents/kWh								
Probability distribution	Uniform			Uniform				
Limiting values	0.1	0.2		0.15	0.25			
Discount rate								
Probability distribution	Uniform			Uniform				
Limiting values	0.05	0.08		0.05	0.08			
Fuel cost increase rate								
Probability distribution				Uniform				
Limiting values, best estimate				0.03	0.05			
Limiting values, optimistic				0.02	0.04			

5. RESULTS OF CALCULATIONS

Levelized cost of produced electrical energy in coal, natural gas and nuclear power plants based of best estimated input data given in Tables I and II are shown in Figure 1. It cold be seen that under specified circumstances (primarily due to forecast gas costs increase rate) cost of electrical power produced in nuclear power plant could be lower than produced by competitive technologies. No external costs are taken into account in this calculations. These costs would increase advantage of nuclear power.

In order to explore the role of electrical energy production with renewable sources to the economics of electric power generation two cases were compared:

— Cost of electrical energy produced by combined cycle natural gas plant only operating with a load factor as required by the consumers
— Cost to produce the same amount of energy by combined generators consisting wind power plants operating with their potential load factor (20-30%) and combined cycle gas plants.

It has to be remembered that avoided system costs caused by renewable generators are higher when system variable costs (which are mainly fuel costs) are higher. Therefore the economics of renewable power is less when the system consists of coal and specially of nuclear power plants. The main reason for introducing renewable energy sources for electricity generation is their favourable environmental impact. It is therefore normal that in comparing electricity costs of renewable and fossil plants to include in calculation also respective external costs. External costs applied in calculation are based on EU ExtenE study in which external costs for gas plants are specified between 1.8 and 2.6 Us cents/kWh (mainly due to carbon and nitrogen oxide emissions) and for wind plants between 0.1 and 0.26 US cents/kWh. Best estimate gas costs are applied with no additional cost increase. By using the cost data from Table II the results are shown in Figure 3.

The figure shows that system containing only gas units produces electricity with lower cost, even if external costs are taken into account. This could be true for best estimate gas costs at the end of this decade. Further increase of gas cost my revert the picture.

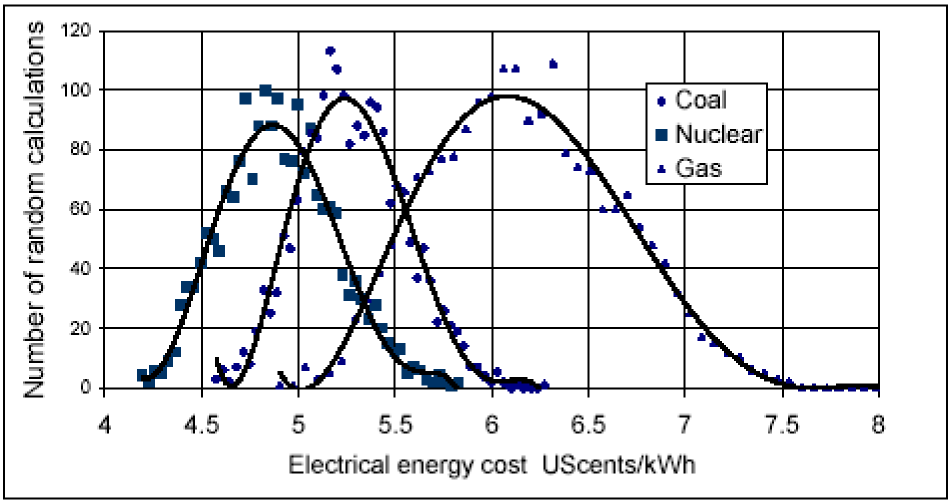

FIG. 2. Levelized electrical energy plant live time cost produced in coal, natural gas and nuclear plants.

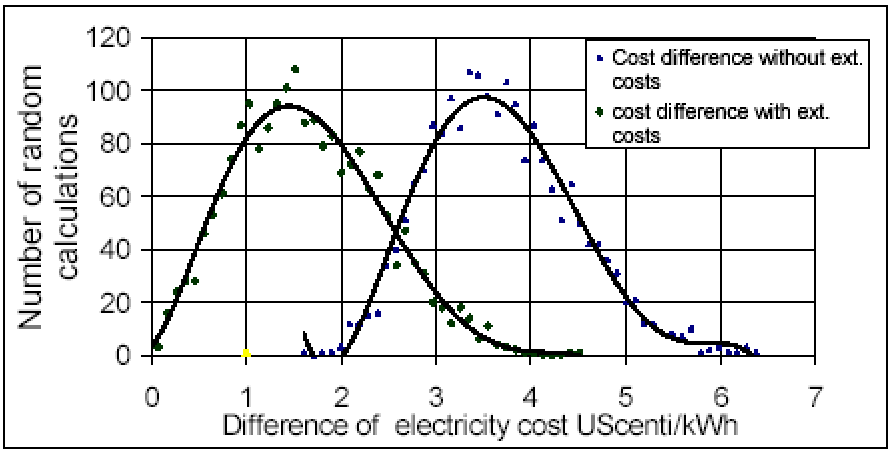

FIG. 3. Difference of produced electricity costs in generating system consisting of natural gas and wind plants and of gas plants only with and without external cost.

5.1. Competitive NPP investment costs

In order to explore limiting specific investments cost of future NPPs (built around the year 2010) the expression (1) is used to calculate the nuclear plant investment cost which would make nuclear plant competitive with fossil plants and plants using renewable energy sources. Presently most economical and also most acceptable conventional plants for environmental impact are combined cycle natural gas plants and wind power plants. These plants are chosen to determine competitive limits of specific investment costs for nuclear power plants. Since external cost are not widely accepted as a factor in economic studies for power generation no external costs are included in calculations This fact will lower the competitive specific investment costs of nuclear plants when compared with natural gas plants. In order to secure a conservative approach to this problem optimistic input data were used for gas and wind plant investment, fuel cost and fuel cost increase rate. On the other hand no optimistic assumption for nuclear power plants are applied. Fixed annual operational and maintenance costs (without fuel) for future nuclear plants are taken between 100 and 120 US$/kW, which is higher than in most presently operating plants.

The results of calculation are given in Figures 4., 5. and 6.

It can be seen from Figures 4. and 5, that medium values of competitive specific investment costs of nuclear power plants when compared with a electricity generating system consisting of gas and wind units and gas units only are around 1900 and 1700 US$/kW respectively.

Figure 6 shows that, due to optimistic assumption of gas cost at the end of this decade and of further gas cost increase, the alternative including only natural cycle gas fired requires a lower competitive NPP specific investment (the distribution showed in Fig. 6 gives a most probable difference of about 200 US$/kW).With best estimated gas cost the option including wind power could prevail.

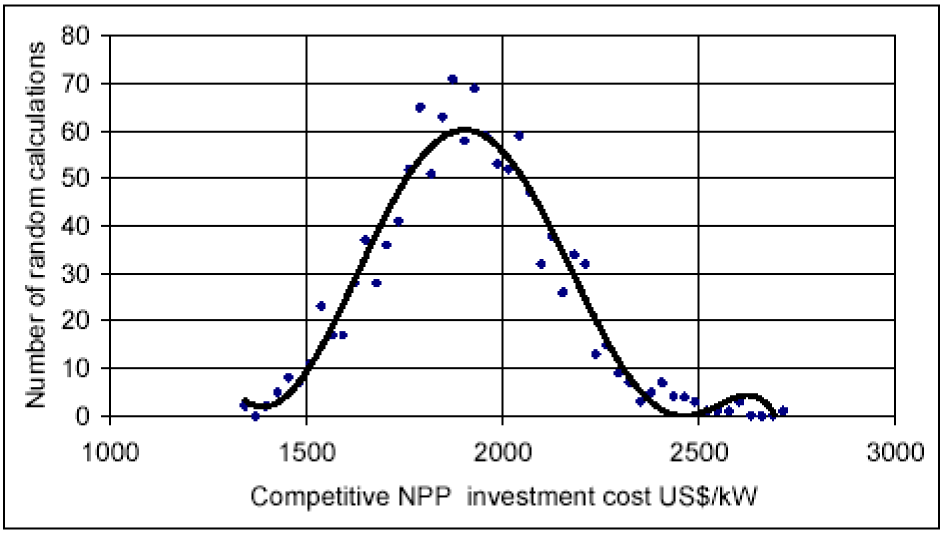

FIG. 4. Distribution of NPP competitive specific investment cost when compared with a electricity generating system consisting of combined cycle natural gas plants and wind electricity generators.

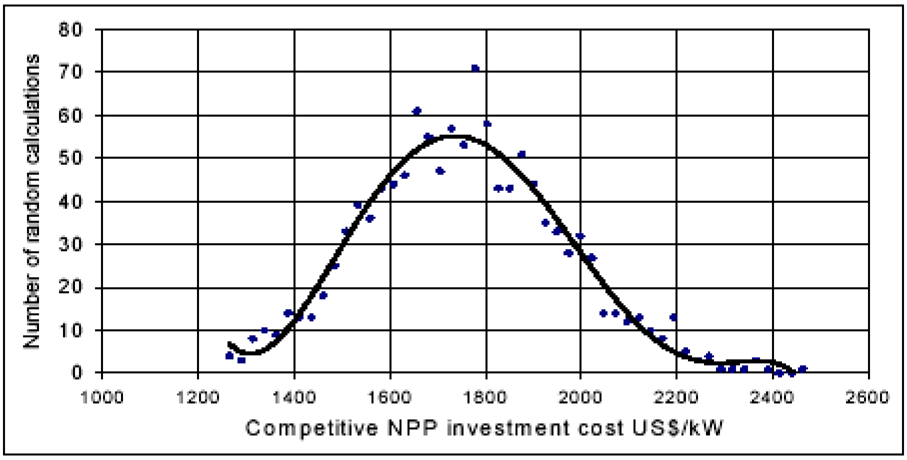

FIG. 5. Distribution of NPP competitive specific investment cost when compared with combined cycle natural gas plants only.

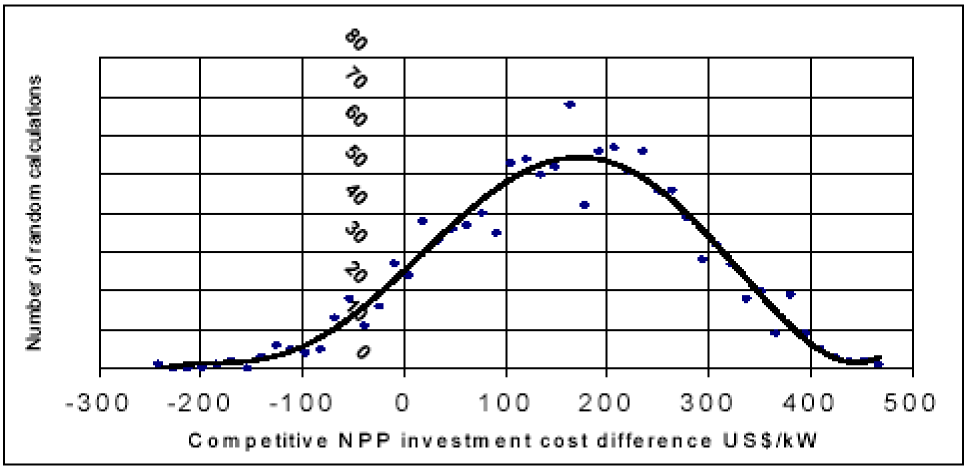

FIG. 6. Difference of nuclear power plant competitive investment costs in Fig. 4 and 5.

6. CONCLUSIONS

The presented analysis illustrates the applicability of probabilistic method to compare the economics of electrical power generating system. The calculation has shown that under the described assumptions competitively of future nuclear power plants could be achieved with their specific investment costs which are not significantly lower than the costs of present units. This fact indicates that the main effort in future innovative reactor design should more concentrated to improve their operational safety and reliability than to substantially lower their investment costs.

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APPROACHES TO THE CREATION OF A CLOSED NUCLEAR CYCLE IN THE RUSSIAN FEDERATION

YU. SOKOLOV, V. RACHKOV

Minatom of Russia, Russian Federation

Abstract. The present basis for nuclear power utilization in the Russian Federation and the country's strategy for future development of nuclear power are discussed in the paper. A closed nuclear fuel cycle with fast reactors is considered essential to make nuclear power utilization sustainable over the long term. Transmutation provides the technological support for prevention of proliferation and it is emphasized that development of such a closed nuclear fuel cycle would be suitable for international cooperation.

1. INTRODUCTION

A number of documents determine at present a basis for the progress of nuclear power in Russia (Figure 1), among them the "Strategy of Russia's nuclear power development in the 1st half of the 21st century".

A closed nuclear fuel cycle (Fig. 2) is determined in that Strategy as the basic trend of the above branch development. Specialists stressed urgency of the given problem since the outset of the nuclear power engineering. There are certain achievements in the said field, but much has to be done (Fig. 3). The basic stages of a transition to a closed nuclear fuel cycle are as follows:

— up to 2010: transition to long-term dry storage, modernization of the RT-1 plant, selection of a radiochemical reprocessing technology for the future RT-2 plant, investigation of minor actinides transmutation in fast reactors;
— after 2010: construction of the RT-2 plant, fabrication of fuel for fast reactors of a new generation, finalizing of elements of the transmutation nuclear fuel cycle (NFC)

FIG. 1. Basis for the progress of nuclear power.

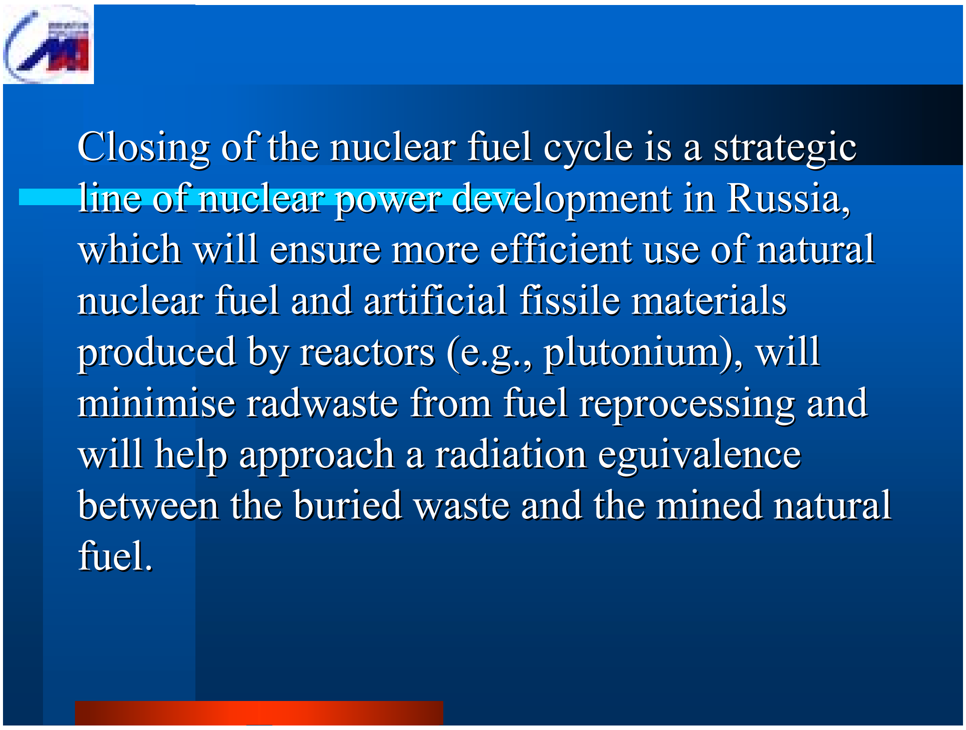


FIG. 2. Strategy of nuclear power development in Russia in the first half of the 21st centur.

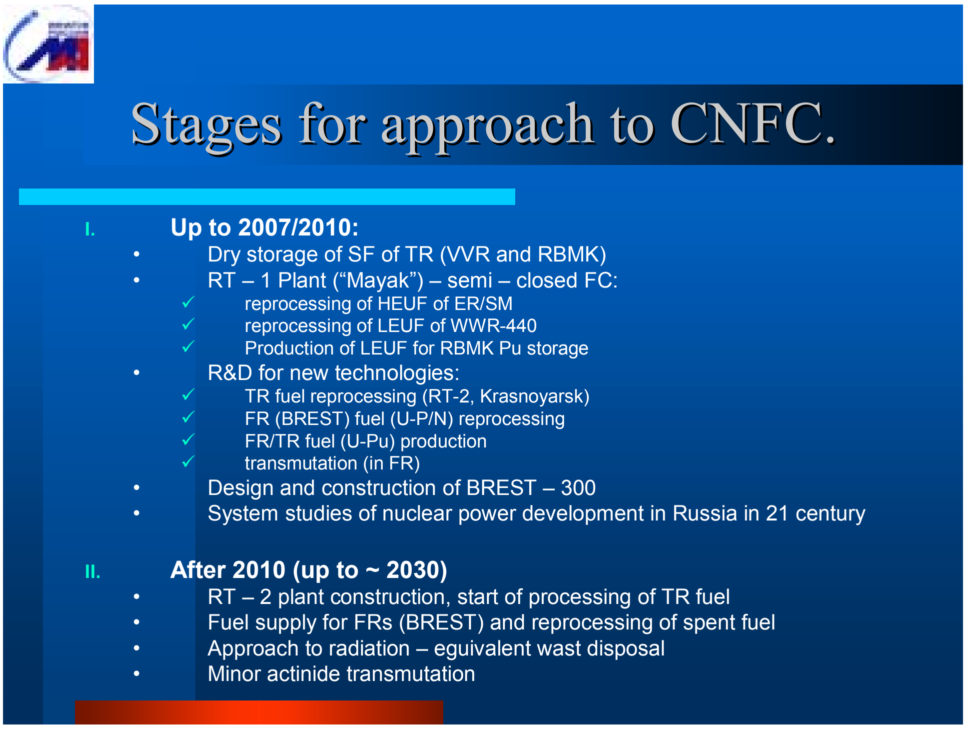


FIG. 3. Strategies for approach to CNFC.

Let us consider a conventional model of an ideal nuclear fuel cycle (Fig. 1.). In that model, fuel nuclides (U and Pu) are returned to the fuel cycle after spent nuclear fuel (SNF) reprocessing, whereas radwastes (RW), removed and placed into geological formations, meet the radiation equivalence principle, enabling RW disposal only after achieving the balance between biological hazard of RW being removed (G_{rw}) and biological hazard of a *corresponding amount* of fuel (G_f).

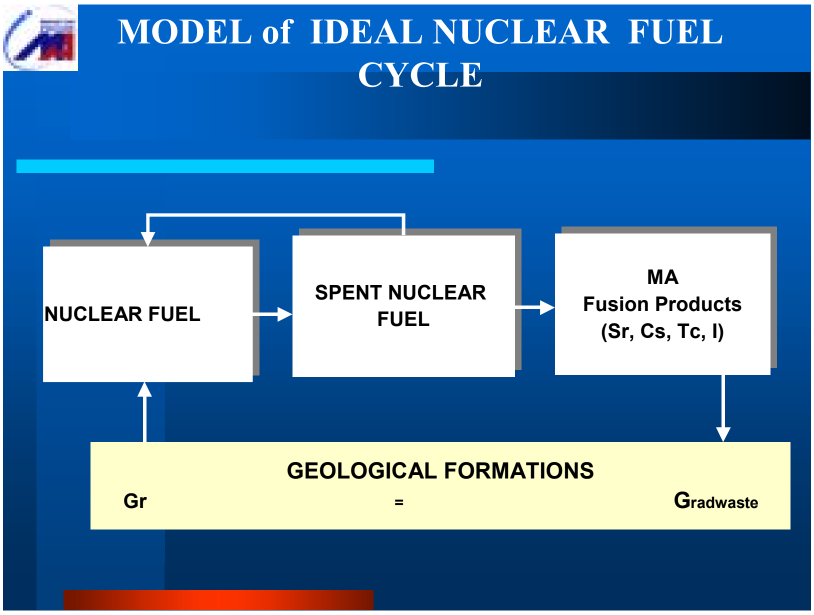


FIG. 4. Model of ideal nuclear fuel cycle.

An important element of the said model is the principle of radiation equivalence (Fig. 5), provided it is considered not as a process but as an instrument for specifying quantitative requirements for a radiochemical composition of RW. Quantitative requirements for a radiochemical composition of RW give birth to requirements for a depth of SNF reprocessing, and the following values can be considered as the first approach to the latter: 0.1% U, Pu, MA (minor actinides); 1% Sr, Cs, Tc, I; almost 100% of FP (fission products).

Let us return to the nuclear fuel cycle model (Fig. 6).

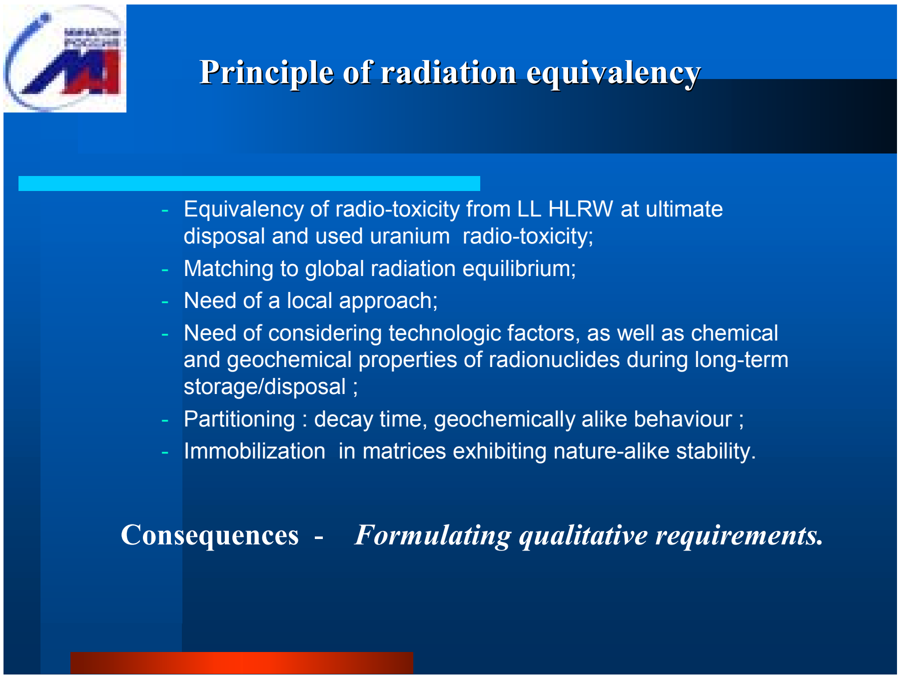


FIG. 5. Princile of radiation equivalency.

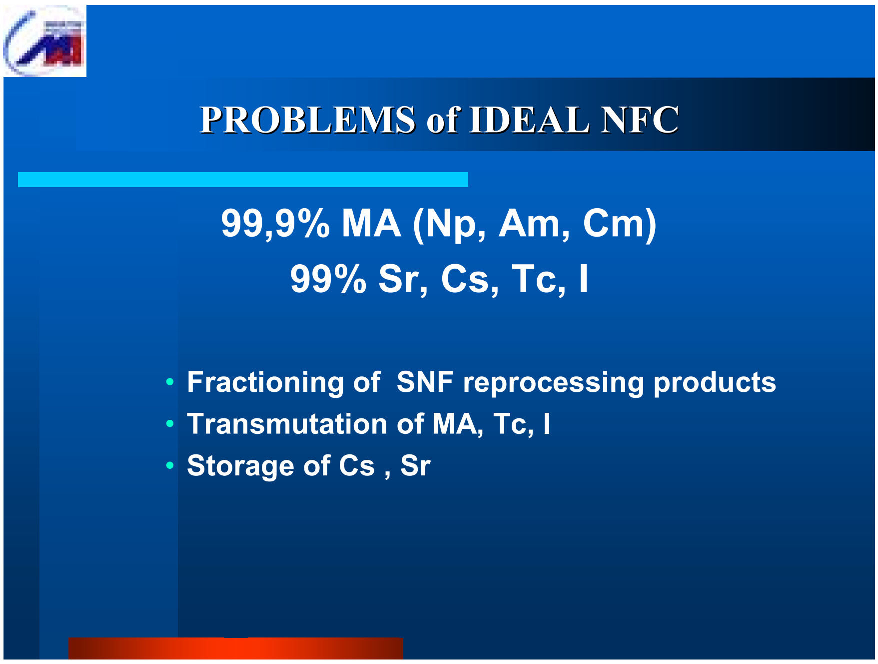


FIG. 6. Problems of ideal NFC.

Now we can face such a problem as management of 99.9% of MA and 99% of Sr, Cs, Tc, I. The following tasks shall be solved here:

— fractionation of SNF reprocessing products;
— transmutation of MA, Tc, I;
— storage of Cs and Sr.

Prior to passing to an estimate of possible solutions, let us consider an actual state of the elements of the nuclear fuel cycle (NFC). As it is known, any SNF incorporates the following elements: SNF management, including storage and reprocessing of SNF; and RW management, including RW immobilization, storage, and disposal.

Let us start with SNF management. The table (Fig. 7) presents the current state of the NFC elements and trends of their development with the available preliminary results. With regard to storage of SNF, nowadays, use is made of "wet" storage. However, long-term "wet" storage a leads to corrosion of fuel assemblies. Because of that, a decision was taken to pass over to long-term "dry" storage, being partially realized at the Leningrad and Bilibino NPPs. The Program concerning management of nuclear fuel from power, transport, and research reactors for facilitating the creation of a closed nuclear fuel cycle in Russia envisages as follows:

— increased (to 9,000 t) capacity of the acting storage facility for SNF from the VVER-1000 reactors at the Mining & Chemical Combine (MCC);
— construction of a "dry" storage facility of the 33,000 t capacity for SNF from VVER-1000 and RBMK-1000 reactors at MCC;
— completion of the construction of a storage facility for SNF from nuclear-powered submarines (NPS) and NPPs with VVER-1000 reactors at the Mayak Production Association (PA).

Radiochemical reprocessing of spent fuel, carried out for over twenty years at the Mayak PA, will continue after its modernization. In future, spent fuel will be also reprocessed at the RT-2 plant (MCC). Pursuant to the "Strategy of Russia's nuclear power development in the 1st half of the 21st century", it is recommended that reprocessing of the bulk of the SNF should be postponed until starting a lot production of fast reactors of a new generation.

At present, the RT-1 plant reprocesses SNF from VVER-440, BN-600, transport, and research reactors. The plant's annual design capacity in terms of spent fuel from VVER-440 reactors makes up 400 tons. During reprocessing, use is made of a water extraction technology. Regenerated uranium is applied for fabricating fuel for RBMK-1000 and BN-600 reactors; plutonium dioxide is used for producing ^{238}Pu isotope in reactors; after regeneration plutonium dioxide is placed into storage. Reprocessing of SNF is accompanied with the formation of liquid RW being dispatched for solidification.

A technology for SNF regeneration, being elaborated especially for the RT-1 plant (Fig. 8) subject to modernization, retains the basic element of the currently used technology – the extraction cycle with tributylphosphate as an extractive agent. Being completed with separation mass-exchange operations, an extraction cycle with the minimized (by volume) feed flow and intra-cycling evaporative operations shall solve the following tasks:

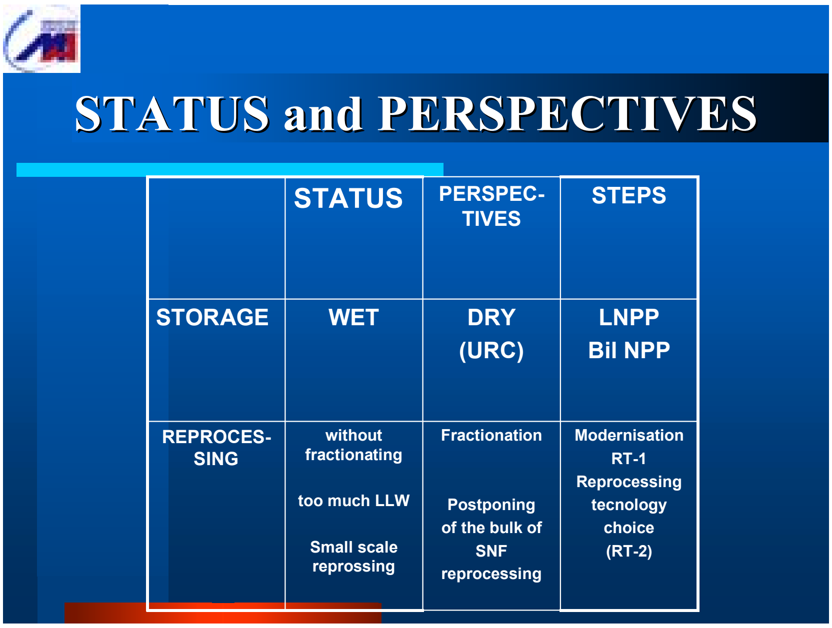


FIG. 7. Current state of the NFC elements and trends.

- — obtaining an uranium product as an uranile nitrate solution;
- — obtaining a plutonium product;
- — obtaining Cs-Sr and MA-REE (rare-earth elements) fractions.
- — The main disadvantages of the present-day system for management of low-level liquid radwastes (LRW) of radiochemical and chemical facilities of the Mayak PA are as follows (Fig. 9):
- — large amounts of initial LRW;
- — discharge of secondary RW as solutions and pulps into open basins of the Techensky cascade;
- — large amounts of secondary wastes;
- — use of process operations leading to considerable salinization of initial LRW;
- — insufficient purification from some nuclides.

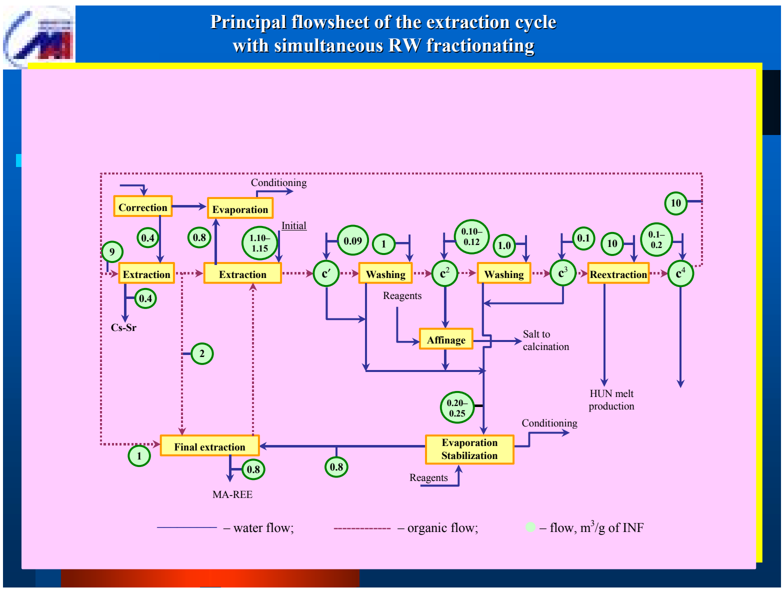


FIG. 8. Technology for SNF regeneration.

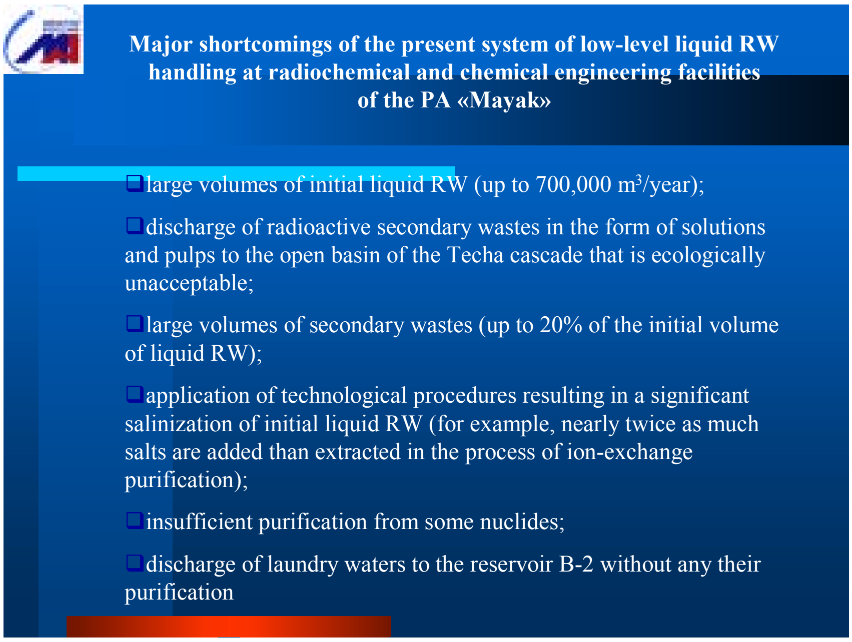


FIG. 9. main disadvantages of the present-day.

The main trends for improving the existing management of low-level LRW at the Mayak PA are as follows (Fig. 10) reduced volume and decreased salinization of LRW by replacing settling and sorption operations by low-reagent methods; and assurance of more effective purification, provision for conditioning of LRW concentrates with complete termination of discharges of radioactive water and pulp into open basins. In association with the Research Institute of Inorganic Materials and other organizations, the Mayak PA has elaborated approaches aimed at improvement of the acting process flow sheets and elimination of the aforementioned shortcomings.

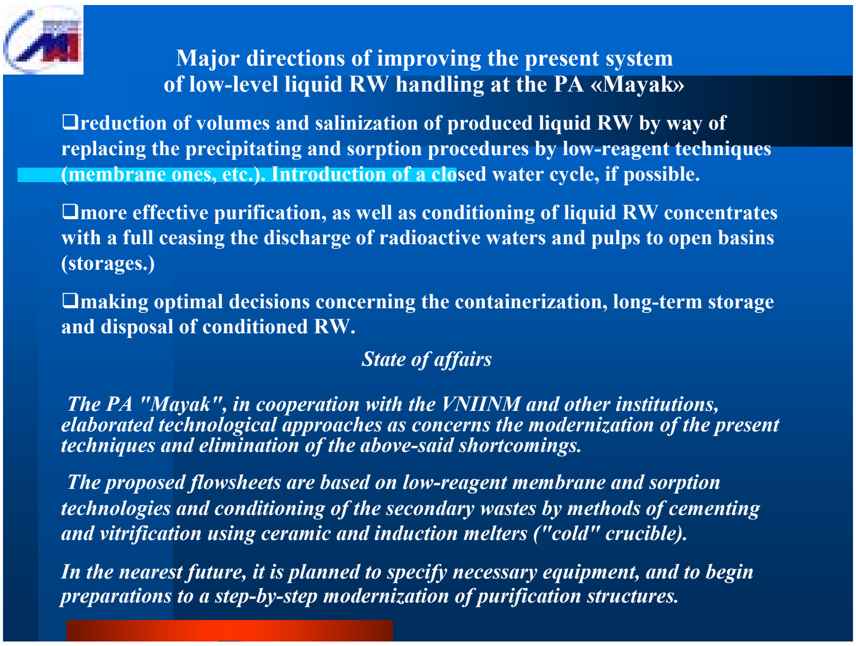


FIG. 10. Main trends for improving.

The proposed flow sheets are based on low-reagent membrane-sorption technologies (Fig. 11). In the nearest future, it is expected to select the required equipment and start getting ready for step-by-step modernization of treatment plants.

The Russian program for creating a closed nuclear fuel cycle envisages construction of the RT-2 plant at MCC with the fabrication of MOX-fuel, as well as erection of a complex for disposal of high-level wastes formed during reprocessing of SNF. After commissioning, the RT-2 plant shall start reprocessing SNF from VVER-1000 reactors (being accumulated now at storage facilities of MCC), foreign reactors PWR and BWR, and, possibly, from Russian reactors of the RBMK type. Both conditioned and defective fuel assemblies will be subject to reprocessing. Several process flow sheets are taken as options for the RT-2 plant, a list of options being given on Fig. 12.

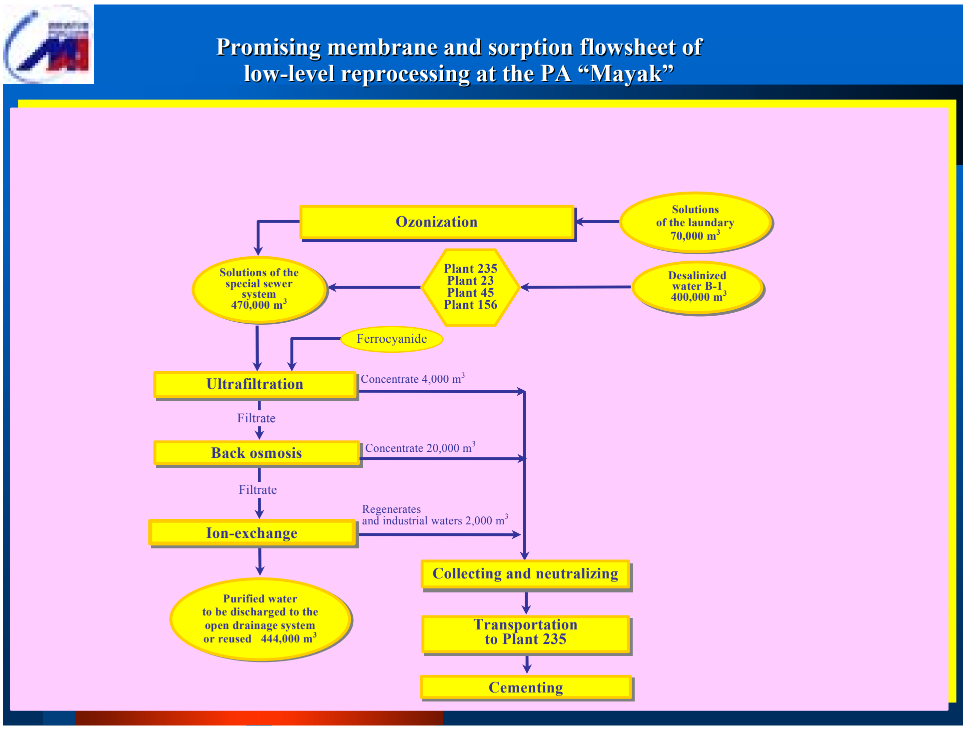


FIG. 11. Low-reagent membrane-sorption technologies.

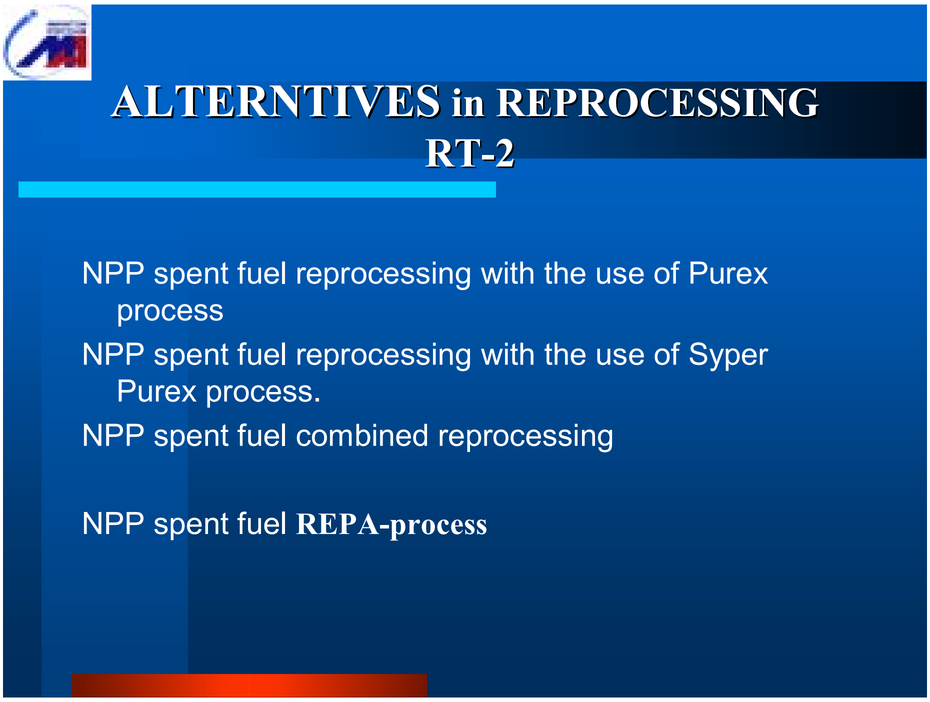


FIG. 12. Several process flow sheets options for the RT-2 plant.

Figure 13 gives a conventional flow sheet for SNF reprocessing on the basis of PUREX, a process with the full separation of U and Pu. As it is known, the said technology has been introduced on an industrial scale. Figure 14 gives a flow sheet for SNF reprocessing with the use of the so-called Super PUREX process without separation of U and Pu, which are used for fabricating fuel for promising fast reactors. In this year, this technology has been tested in hot chambers of the Institute of Radium.

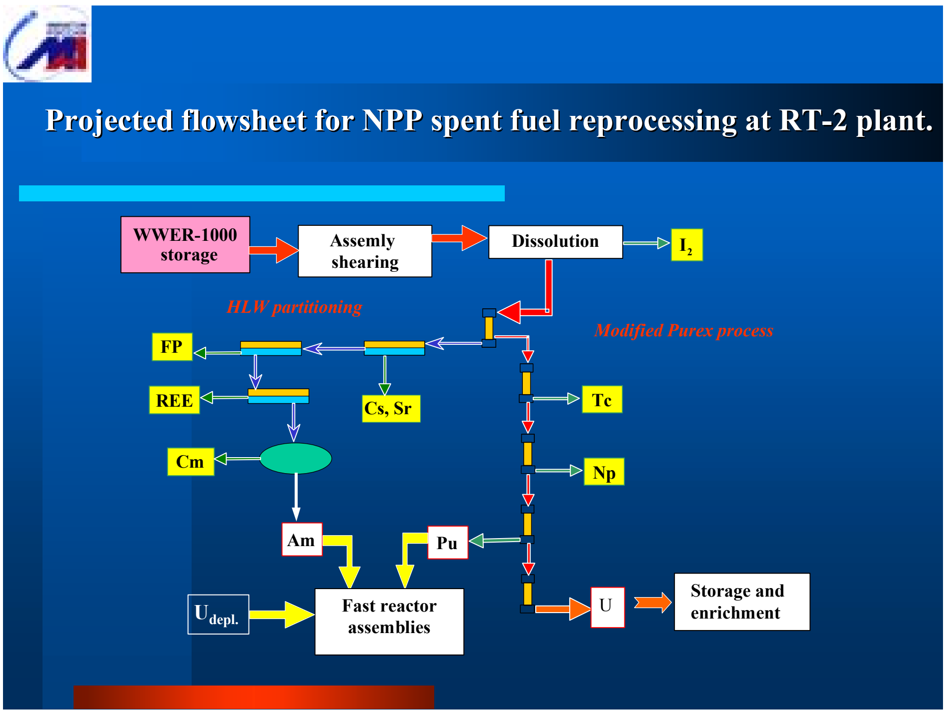


FIG. 13. Conventional flow sheet for SNF reprocessing on the basis of PUREX.

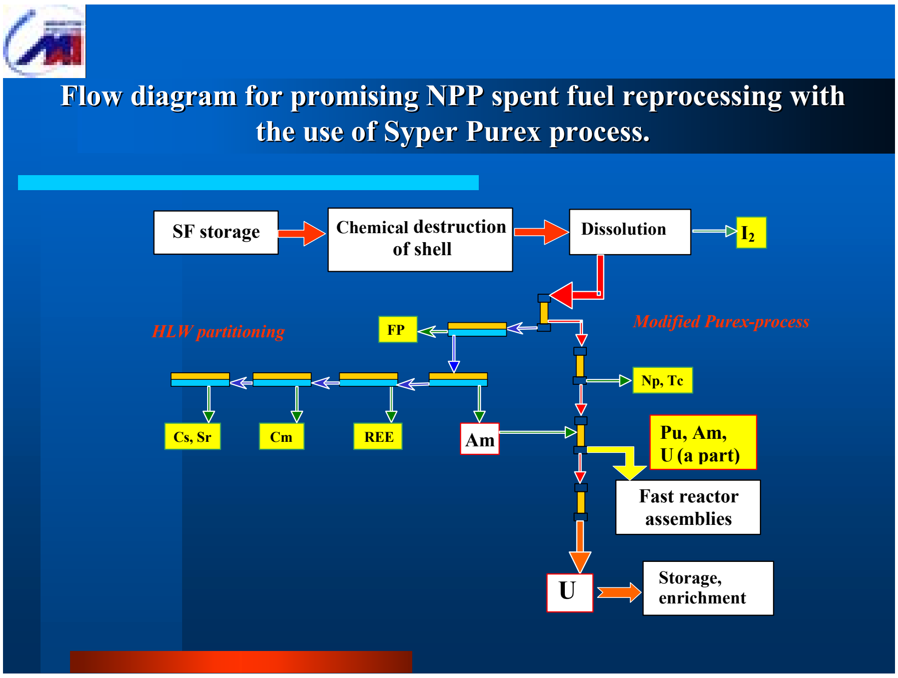


FIG. 14. Flow sheet for SNF reprocessing.

Slide 15 gives a flow sheet for radiochemical reprocessing with the use of the PUREX process (for simultaneous separation of U and Pu) in combination with the gas-cycle fluorination (for separation of the main part of U). Laboratory tests of that technology have been conducted at the Kurchatov Institute).

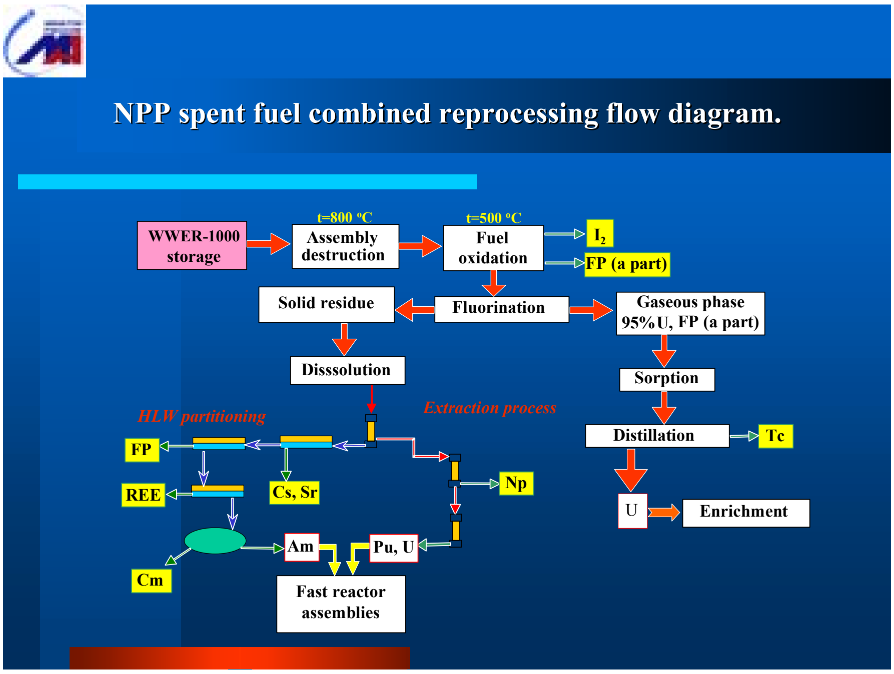


FIG. 15. Flow sheet for radiochemical reprocessing.

Figure 16 gives a flow sheet for radiochemical reprocessing with the use of the PUREX process (for simultaneous separation of U and Pu) in combination with the gas-liquid conversion of uranium oxide in a mixture of carbonic acid and N_2O_4 (for separation of the main part of U). At present, the said technology is tested in hot chambers of MCC. Other technologies are also under investigation.

Comparative characteristics of some flow sheets for the RT-2 plant are given on Figure 17.

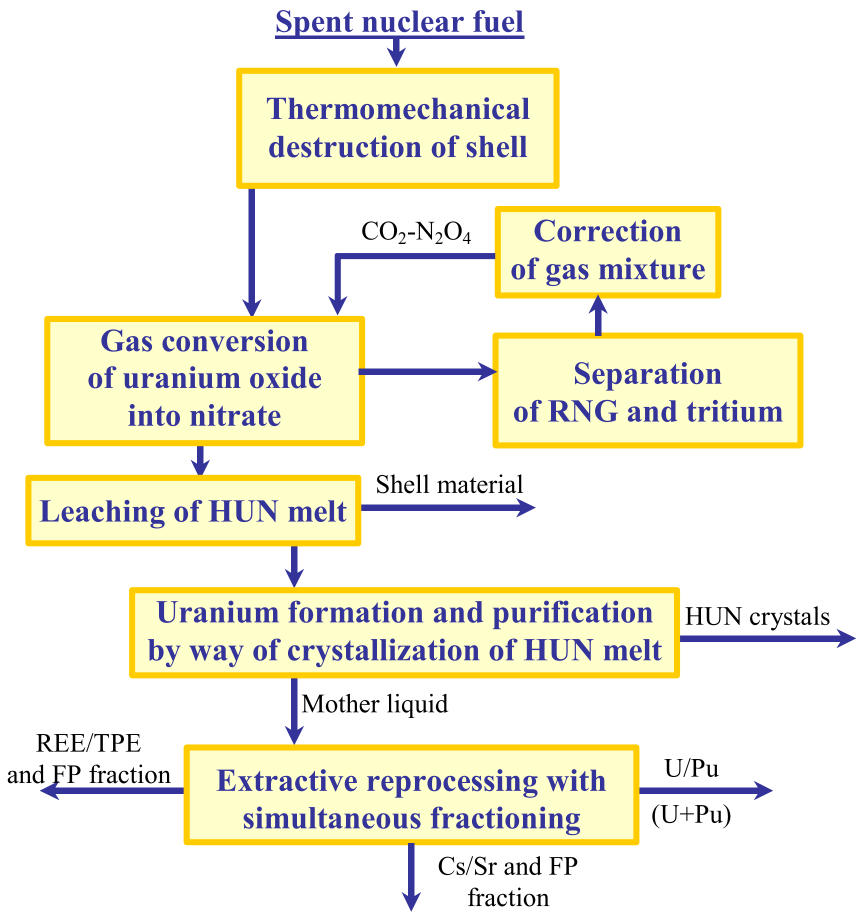
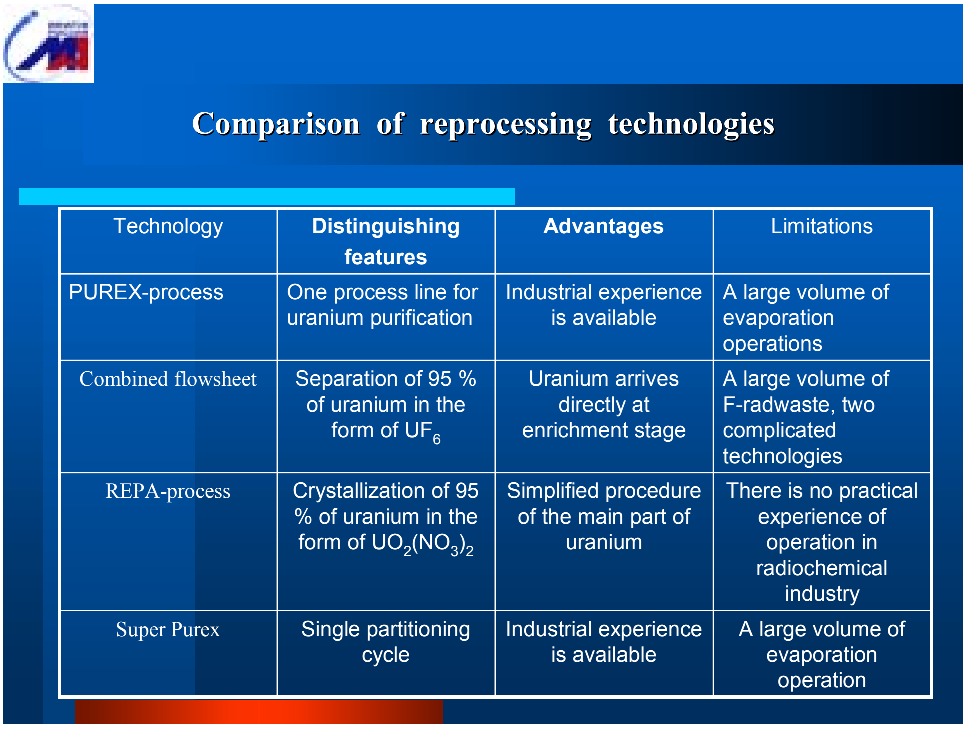


FIG. 16. Flow sheet for radiochemical reprocessing.



Technology	Distinguishing features	Advantages	Limitations
PUREX-process	One process line for uranium purification	Industrial experience is available	A large volume of evaporation operations
Combined flowsheet	Separation of 95 % of uranium in the form of UF_6	Uranium arrives directly at enrichment stage	A large volume of F-radwaste, two complicated technologies
REPA-process	Crystallization of 95 % of uranium in the form of $UO_2(NO_3)_2$	Simplified procedure of the main part of uranium	There is no practical experience of operation in radiochemical industry
Super Purex	Single partitioning cycle	Industrial experience is available	A large volume of evaporation operation

FIG. 17. Comparative characteristics of some flow sheets for the RT-2 plant.

Having considered the present elements of NFC, we shall return to the model of a nuclear fuel cycle, presented in Fig. 18. This model can be treated as the one of a transmutation nuclear fuel cycle.

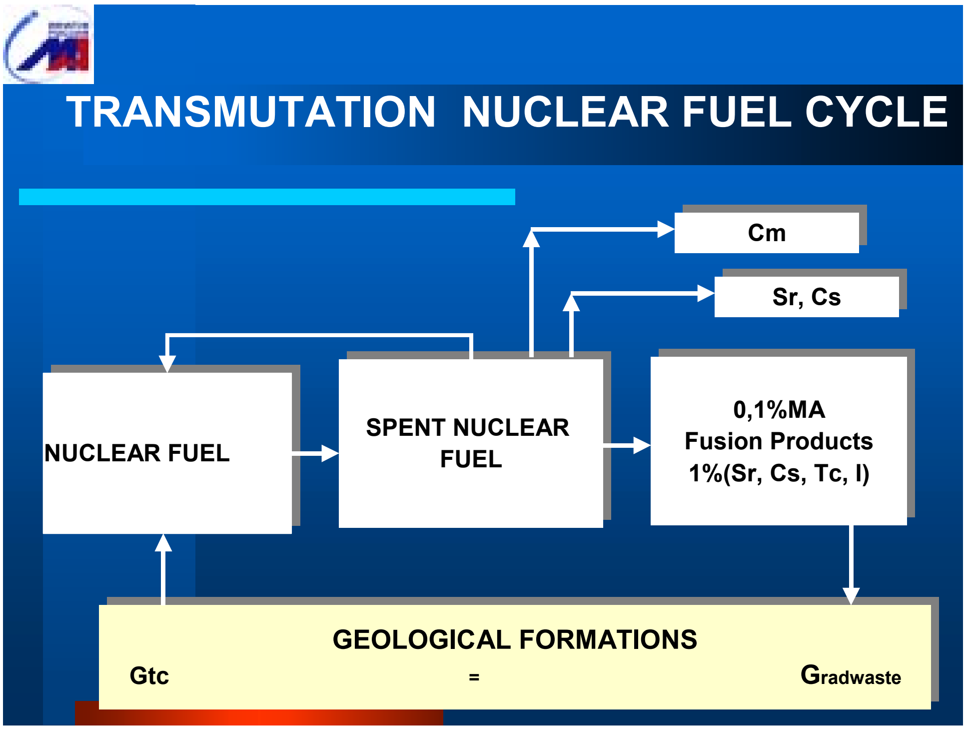


FIG. 18. the model of a nuclear fuel cycle.

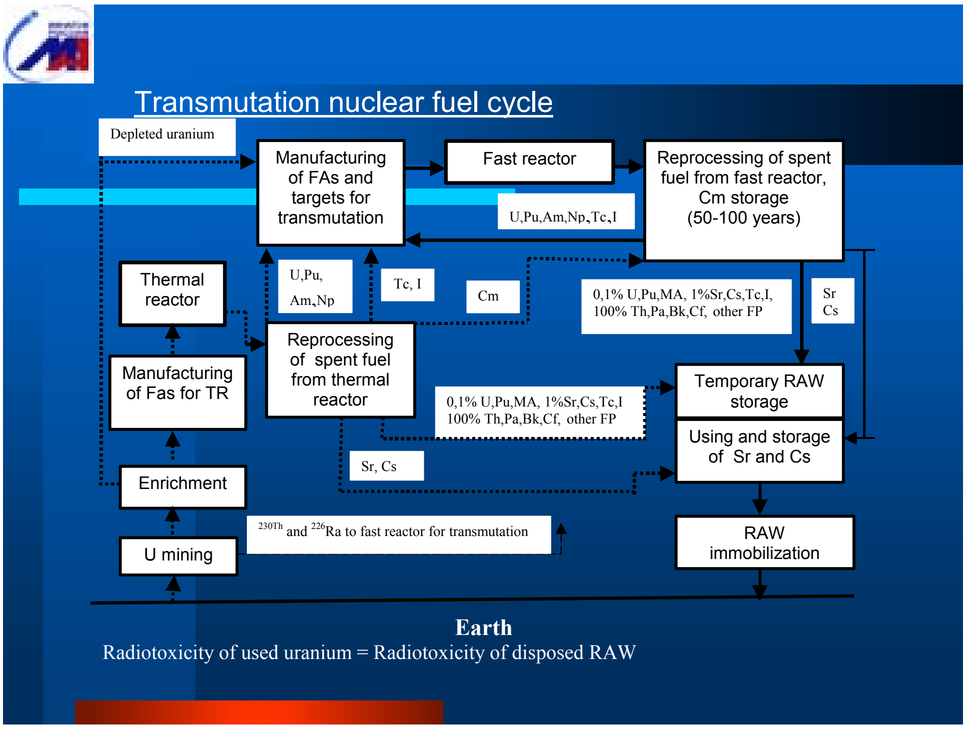


FIG. 19. Transmutation of the nuclear fuel cycle.

Figure 19 gives a detailed flow sheet of the above NFC model.

The said cycle is based on the following:

— reprocessing of SNF with the minimal possible residue of U, Pu, minor actinides, and long-lived high-level fission products;
— transmutation (in reactors) of minor actinides (Am, Cm, possibly Np and Pa) into fission products;
— transmutation of long-lived high-level fission products **(^{129}I, ^{99}Tc)** into a stable state;
— utilization of **^{90}Sr, ^{137}Cs**, and several other elements;
— long-term controlled cooling of long-lived high-level wastes prior to their final disposal in spent and reclaimed uranium mines or geological formations.

Transition from the conventional to the transmutation NFC can bring to life the following problems:

— fabrication and working ability of "dirty" nuclear fuel, i.e. fuel with inclusions of transmutation nuclides;
— reprocessing of SNF from fast reactors with "dirty" fuel.
— One of the promising technologies for fabricating "dirty" nuclear fuel is a vibration technology being elaborated and tested at the Research Institute of Nuclear Reactors – RINR). The primary features of the given technology are as follows (Fig. 20):
    - simplicity and reliability of the process owing to a reduced number of operations and checking procedures, facilitating full automation of the process;

- possibility of using granules of initial materials for forming both a homogeneous composition and a mechanical mixture;
- reduced thermomechanical impact of vibro-fuel on structural elements of fuel assemblies;
- more flexible requirements for the inner diameter of a fuel slug.

The results of reactor tests of MOX-fuel fabricated with the aid of the said vibration technology are presented on Fig. 21.

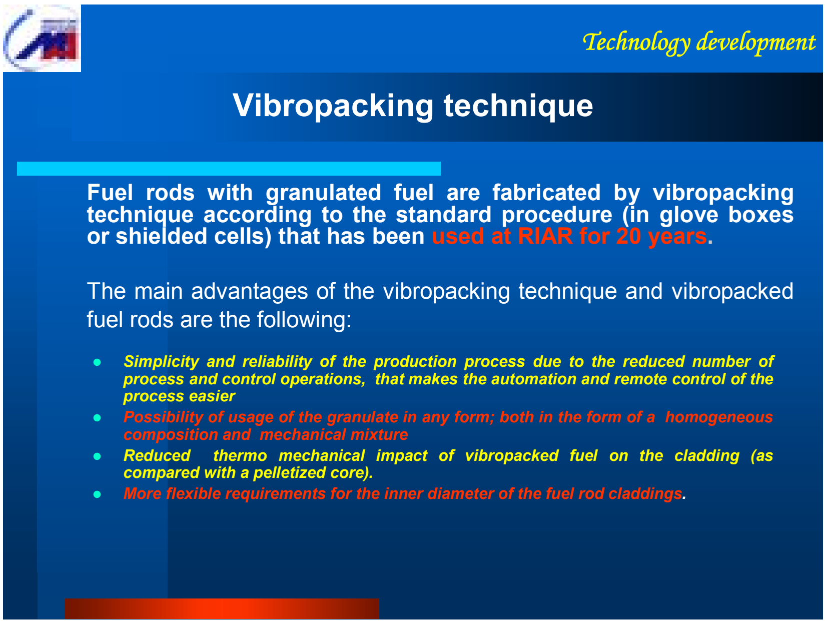


FIG. 20. Primary features.

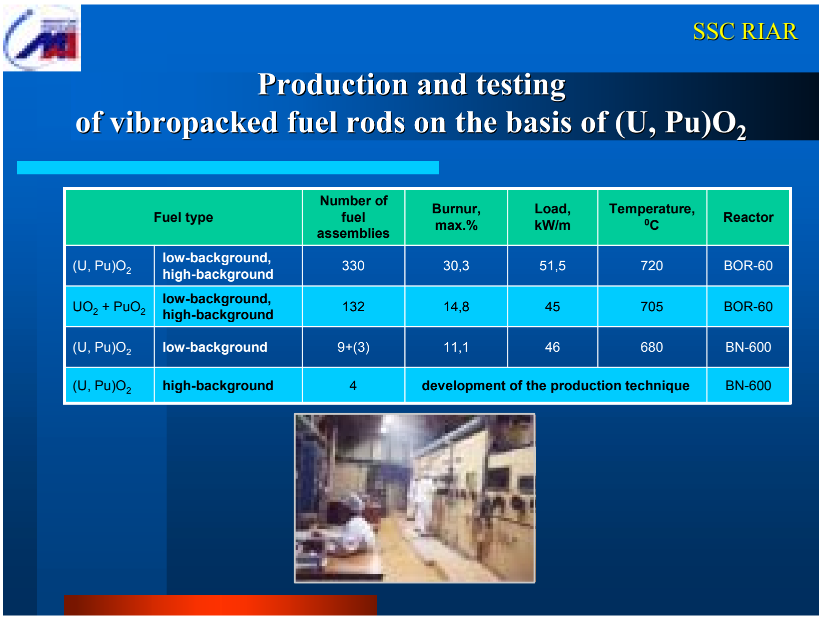


FIG. 21. Reactor tests of MOX-fuel.

Within the framework of the DOVITA program a batch of granulated uranium-neptunium oxide fuel with neptunium content of 5 per cent was produced

✓ **On the basis of the fuel produced, vibro-packed fuel rods with a fast reactor cladding were fabricated using the equipment designed for BOR-60 fuel rods fabrication.**

The isotopic content of fuel rods with neptunium fuel
(irradiation during **694** days in BOR-60 reactor)

Isotope	Mass, g		Changes in isotope mass
	Before irradiation	After irradiation	
^{235}U	53.5	42.9	- 10.6
^{238}U	16.5	15.96	- 0.54
^{237}Np	4.63	3.93	- 0.70
^{234}U	-	0.012	0.012
^{236}U	-	1.69	1.690
^{238}Pu	-	0.34	0.340
^{239}Pu	-	0.31	0.310
^{240}Pu	-	$4.2*10^{-3}$	$4.2*10^{-3}$
FP	-	9.48	9.480

At present, 20 per cent of the burnup is achieved in the fuel rods with neptunium fuel

Experimental fuel rods with $(U,Pu,Am)O_2$ fuel, containing 3 per cent of americium, have been prepared, and in the year 2002 they will be subjected to irradiation in the BOR-60 reactor

FIG. 22. Experience in introduction of minor-actinides into the fuel composition and irradiation results. The results of over 700-day tests of oxide vibro-fuel with a 5-% content of **Np** in the BOR-60 reactor are given on Fig. 22.

Nowadays RINR is carrying out reactor tests of vibro MOX-fuel with a 3-% content of **Am**. The vibration technology is considered as one of techniques for fabricating nitride nuclear fuel (Fig. 23).

Properties	$UPuO_2$	UPuN
Density (g/cm^3)	11,05	14,32
Contains of fissile materials in 1 cm^3 fuel (g/cm^3)	9,74	13,53
Thermal conductivity by temperature 500-1000^{0}C (wt/mK)	2,2-2,0	20-22
Temperature of melting (K)	2950	3050
Interaction with thermal carriers: Na, Pb, Pb-Bi	Interection with Na and (Na_3UPuO_4+Q) creation	Not interaction

FIG. 23. Properties of oxide and nitride

fuel.

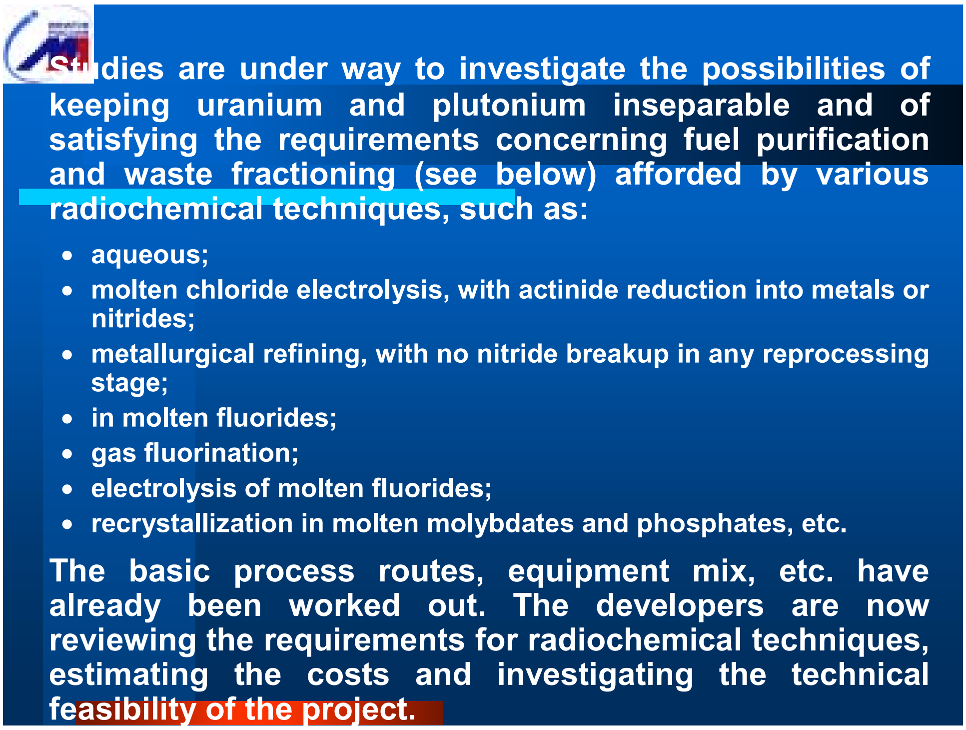


FIG. 24. Alternatives in FR fuel reprocessing.

A growing interest in the fuel with a high density and heat conductivity is associated with the design of a fast reactor BREST satisfying the principle of natural safety. For radiochemical reprocessing of SNF from promising fast reactors, which shall become key elements of the transmutation nuclear fuel cycle, the following radiochemical technologies are being considered at present (Fig. 24):

— combined, using the PUREX elements and those of “dry” technologies;
— electrolysis of melts of chlorides with the reestablishment of actinides to metals or nitrides;
— metallurgical, without destruction of nitrides at all reprocessing stages;
— regeneration in fluoride melts;
— gas-fluoride technology;
— electrolysis of fluoride melts;
— recrystallization in molybdate and phosphate melts.

For the above technologies, flow sheets have been elaborated, composition and amount of equipment determined. Work at the key stages has started. Estimation of requirements for those technologies, as well as feasibility studies and evaluation of the production costs are underway. A flow sheet for one of the variants under discussion, i.e. electrolysis in chloride melts, is presented on Figure 25**.**

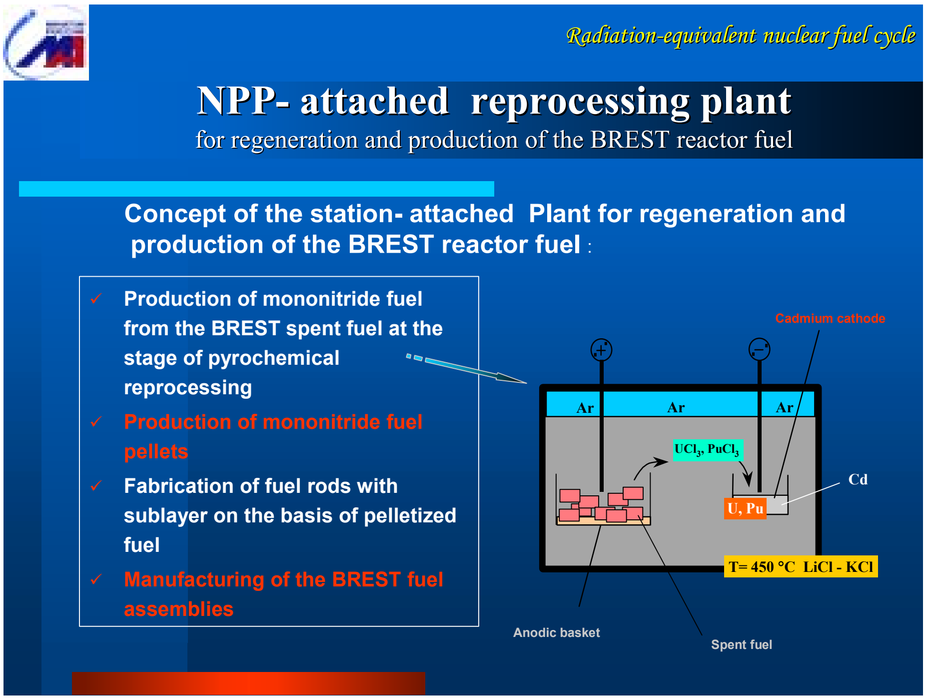


FIG. 25. A flow sheet for electrolysis in melting chloride.

Spent nuclear fuel gets dissolved in the LiCl-KCl eutectic melt and is reestablished to metal (U and Pu) on the cathode. The results of the above variant testing, conducted in hot chambers of RINR and relating to SNF from the BN-350 reactor (UO_2), and SNF from the BOR-60 reactor (UPu O_2 and UO_2) are given on Figure 26**.**

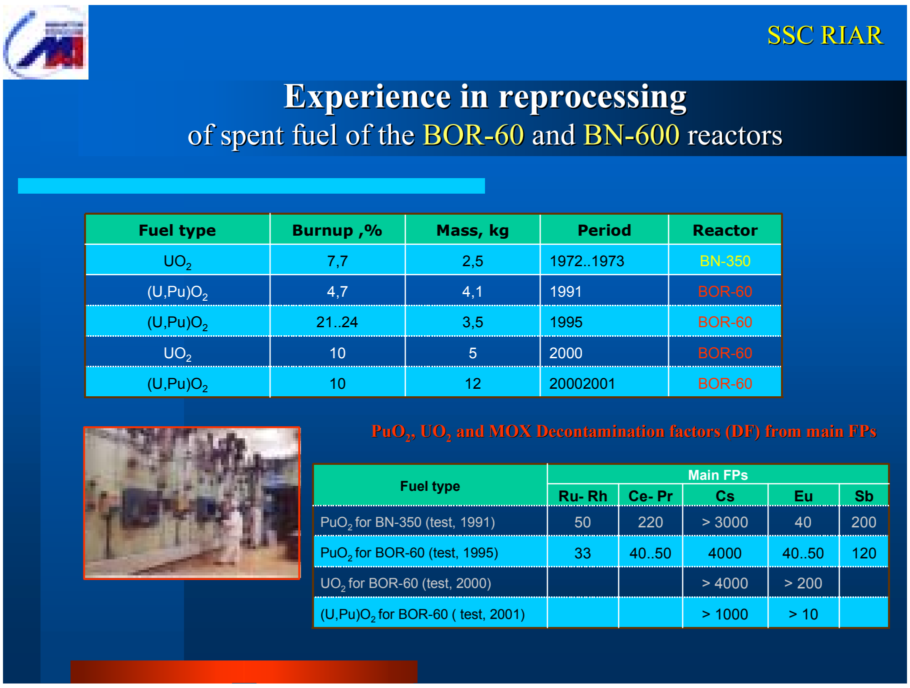


FIG. 26. Results of testing.

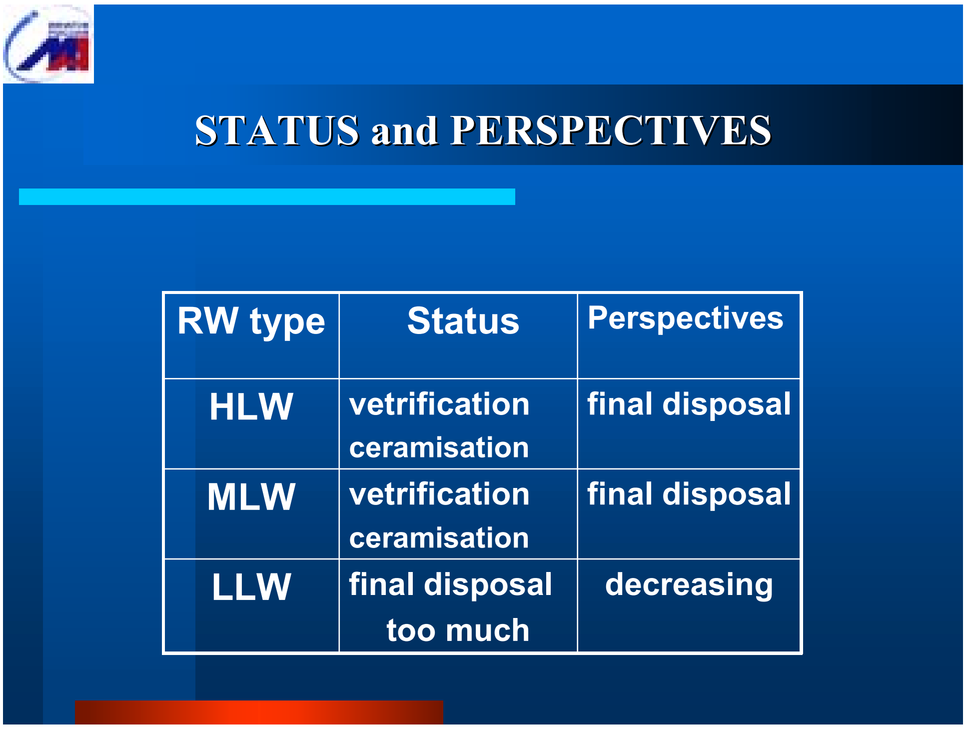


FIG. 27. Current state of the NFC.

Now let us turn to the problems of RW management. The table (Fig. 27) reflects the current state of this element of the NFC and the trend of its development with available intermediate results.

The present-day management of RW, starting form SNF reprocessing at the RT-1 plant, envisages their vitrification **without prior separation of the radionuclide mixture**, temporary storage of vitrified blocks, and their subsequent disposal.

— **LLW** (low-level wastes). Problems concerning LLW have already been considered above, while discussing modernization of the RT-1 plant.

— **HLW** (high-level wastes). At present, vitrification is the basic process for conditioning of all liquid HLW formed during SNF reprocessing. In Russia, use is primarily made of phosphate glass matrices. The EP-500 industrial-scale instillation, acting at the Mayak PA, applies glass based on Na_2O-Al_2O_3-P_2O_5, where phosphorus and aluminium oxides act as glass formers. For hardening of HLW of a complex composition and origin, mineral-like matrices can be used, synthesized in **the induction smelter with the cold crucible**. A pilot plant for HLW vitrification based on the said technology shall be installed at the Mayak PA.

It should be noted that both the matrix and the HLW immobilization technology should correspond to the SNF reprocessing technique applied. Figure 28 presents the results of immobilization of HLW, formed during SNF reprocessing, in the **glass matrix**, by applying the **electrolytic** method **($NaPO_3$-AlF_3-Al_2O_3** matrix) and the **phosphate** method **($Pb(PO_3)_2$-$NaPO_3$** matrix).

SSC RIAR

Experience in disposition of radioactive waste arising from pyrochemical regeneration of spent fuel

Vitrification of HLW resulted from pyrochemical process

Characteristics	Type of high-level wastes		
	Phosphate deposit	Spent salt electrolyte	Phosphate deposit + spent salt electrolyte
Type of glass matrix	$Pb(PO_3)_2$ $NaPO_3$	$NaPO_3$, AlF_3 Al_2O_3	$NaPO_3$, AlF_3 Al_2O_3
Method of introduction into the glass matrix	Vitrification , T=950^{0}C	Vitrification without chlorides conversion , T=950^{0}C	Vitrification without chlorides conversion, T=950^{0}C
Quantity of wastes introduced , %	28	20	36
Leaching rate of ^{137}Cs on 7-th day, g/cm2 * day	$7*10^{-6}$	$7*10^{-6}$	$4*10^{-6}$
Thermal stability , ^{0}C	400	400	400
Radiation resistance	10^7 Gy(for γ and β) 10^{18} α-decay/g		

FIG. 28. Results of immobilization of HLW.

Figure 29 gives the results of immobilization of HLW, formed during SNF reprocessing, in the **ceramic matrix** using for the purpose **electrolytic** (**monazite)** and phosphate **(cosnarite NZP)** methods.

SSC RIAR

Experience in disposition of radioactive waste arising from pyrochemical regeneration of spent fuel

Ceramization of HLW arising from pyrochemical process

Characteristics	Type of high-level wastes	
	Phosphate deposit	Spent salt electrolyte
Type of ceramics	monazite	Cosnarite (NZP)
Method of introduction into ceramics	pressing, calcination, T=850^{0}C	Conversion to NZP from the melt or aqueous solution, pressing, calcination, T=1000^{0}C
Quantity of waste introduced into ceramics, %	100	30..40
Leaching rate of ^{137}Cs on 7-th day, g/cm2 * day	$1*10^{-6}$	$3*10^{-6}$
Thermal stability, ^{0}C	850	1000
Radiation resistance	$5*10^8$ Gy(for γ and β)	10^{19} α- decay/g

FIG. 29. Other results.

It comes natural that all researches, associated with NFC, make sense only if they are conducted in the interests of the current or promising nuclear power technologies. In this connection, one shall not ignore the fact that any power technology may encounter both inner and outer problems (Fig. 30). For a nuclear power technology, like any other power technology, **inner problems** are associated with fuel (solved by commissioning of fast reactors) and wastes (solved within the transmutation nuclear fuel cycle).

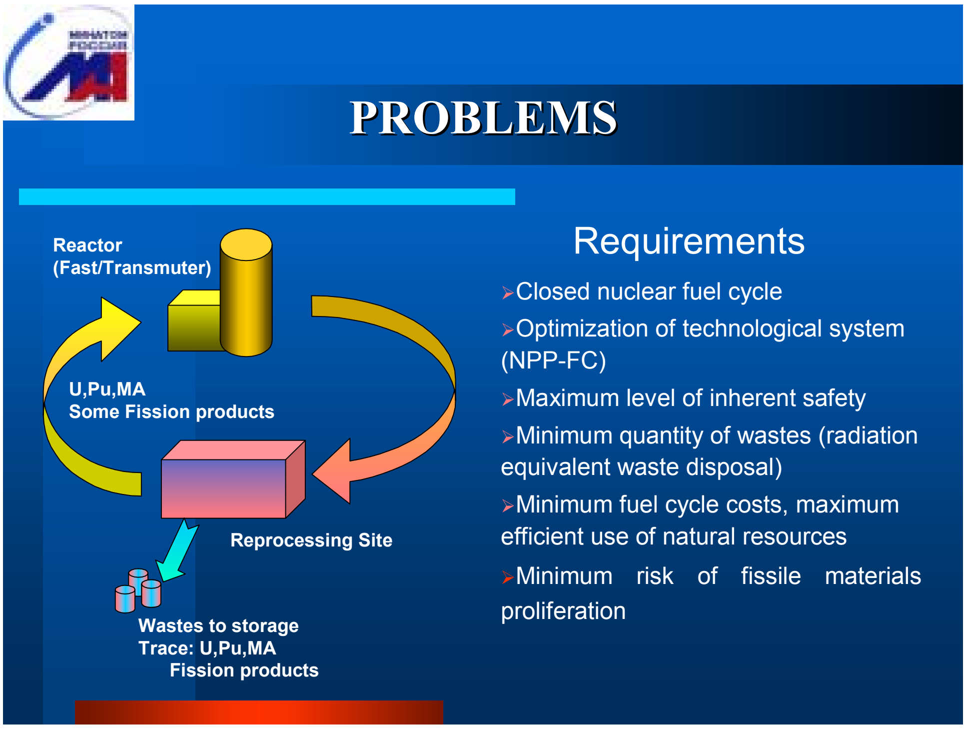


FIG. 30. Challenges.

As regards **outer problems** of nuclear power technologies, they are related to competitiveness and non-proliferation of nuclear weapons. It is evident that while solving inner problems of nuclear power engineering, i.e. choosing this or that variant of a closed nuclear fuel cycle, one shall not ignore outer problems. Progress of wide-scale nuclear power engineering brings to the forefront the observance of the Treaty on non-proliferation of nuclear materials (plutonium, high-enrichment uranium) suitable for making nuclear weapons. This is determined by the present-day political situation, and it is hardly possible that in the nearest 30-50 years the said problem will become less urgent (Fig. 31).

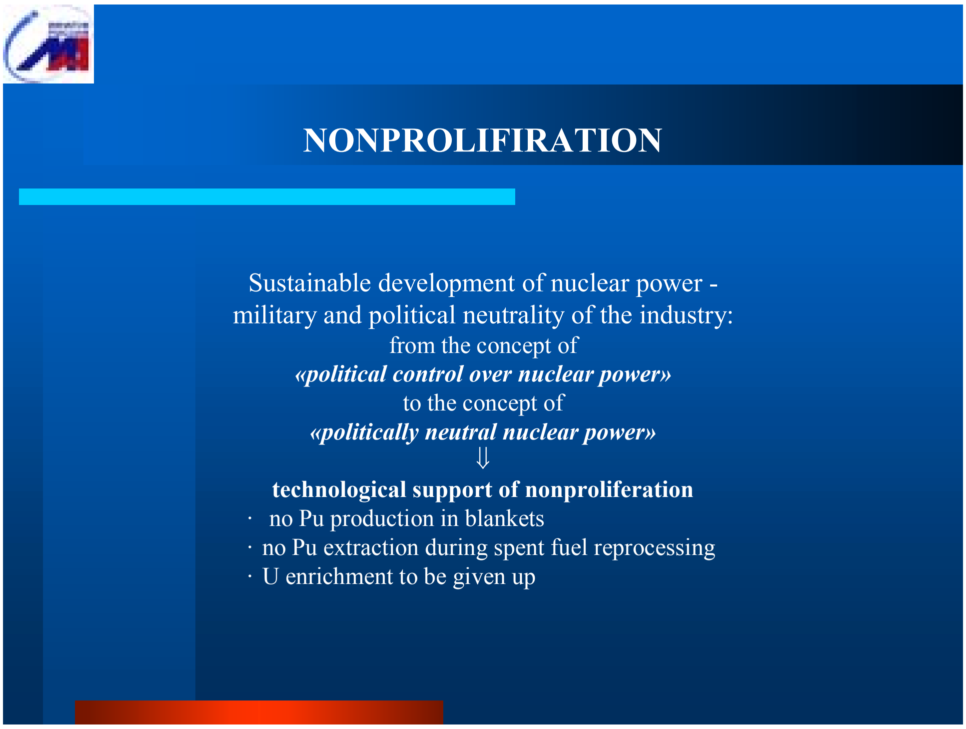


FIG. 31. Nonproliferation challenges.

Various radiochemical technologies, which can be applied to the fuel cycle of a new generation fast reactors, are being investigated nowadays. It is expected that these technologies will be able to provide for non-separation of uranium and plutonium, as well as ensure prescribed requirements for a degree of fuel purification and waste fractionation. Those technologies have been described above.

As concerns economic aspects of nuclear power, the principal task here is elaboration of respective technologies facilitating a transition to the transmutation nuclear power cycle with fast reactors and enhancing competitiveness of national nuclear industry.

2. CONCLUSIONS

1. Closed nuclear fuel cycle is a strategic trend of development of Russia's nuclear power engineering.

2. Large-scale nuclear power engineering means power engineering with fast reactors.

3. Closed nuclear fuel cycle of large-scale nuclear power engineering means transmutation NFC with technological support of the non-proliferation treatment.

4. CNFC problems are suitable for international cooperation.

IAEA PUBLICATIONS RELATED TO THE SUBJECT OF THE CONFERENCE

Publication Number	Title	Year
IAEA-TECODC-936	Terms for Describing New, Advanced Nuclear Power Plants	1997
IAEA-TECDOC-940	Floating Nuclear Energy Plants for Seawater Desalination	1997
IAEA-TECDOC-977	Integral Design Concepts of Advanced Water Cooled Reactors	1997
IAEA-TECDOC-985	Accelerator-Driven Systems: Energy Generation and Transmutation of Nuclear Waste (status report)	1997
IAEA-TC-903.3	Feasibility and Motivation for Hybrid Concepts for Nuclear Energy Generation and Transmutation	1998
IAEA-TECDOC-1015	Advances in Fast Reactor Technology	1998
IAEA-TECDOC-1085	Hydrogen as an Energy Carrier and its Production by Nuclear Power	1999
IAEA-TECDOC-1122	Fuel Cycle Option for LWRs and HWRs	1999
IAEA-TECDOC-1155	Thorium based fuel options for the generation of electricity: Developments in the 1990s	2000
IAEA-TECDOC-1172	Small Power and Head Generation Systems on the Basis of Propulsion and Innovative Reactor Technologies	2000
IAEA-TECDOC-1184	Status of Non-Electric Nuclear Heat Applications: Technology and Safety	2000
IAEA-TECDOC-1198	Current Status and Future Development of Modular High Temperature Gas Cooled Reactor Technology	2001
IAEA-TECDOC-1203	Thermohydraulic Relationships for Advanced Water Cooled Reactors	2001
IAEA-TECDOC-1210	Safety Related Design and Economic Aspects of HTGR	2001
IAEA-TECDOC-1235	Safety Aspects of Nuclear Plants Coupled with Seawater Desalination Units	2001

IAEA-TECDOC-1238	Gas Turbine Power Conversion Systems for Modular HTGRs	2001
IAEA-TECDOC-1264	Reliability Assurance Programme Guidebook for Advanced Light Water Reactors	2001
AEN/NEA and IAEA 2002 Publication	Innovative Nuclear Reactor Development - Opportunities for International Co-operation	2002
IAEA-TECDOC-1245	Performance of Operating and Advanced LWR Design	2002
IAEA-TECDOC-1281	Natural Circulation Data and Methods for Advancedwater cooled nuclear power plant designs	2002
IAEA-TECDOC-1288	Verification of Analysis Methods for Predicting the Behaviour of Seismically Isolated Nuclear Structures	2002
IAEA-TECDOC-1289	Comparative Assessment of Thermopysical and Thermohydraulic Characteristics of Lead, Lead-bismuth and Sodium Coolants	2002
IAEA-TECDOC-1290	Improving Economics and Safety of Water Cooled Reactors: Proven Means and New Approaches	2002
IAEA-TRS-407	HWRs – Status and Projected Development	2002
IAEA-TECDOC-1318	Harmonization and Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes and Relations Using Experimental Data	2002
IAEA-TECDOC-1319	Thorium Fuel Utilization: Options and Trends (Proceeding of three IAEA Meetings held in 1997, 1998 and 1999)	2002
IAEA-TECDOC-1326	Design Concepts of Nuclear Desalination Plants	2002
IAEA-TECDOC-1349	Potential of Thorium-Based Fuel Cycles to Constrain Plutonium and Reduce the Long Lived Waste Toxicity	2003

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	Ms. R. Spiegelberg Planer	NPES
	Ms. K. Morrison	MTCD
	Mr. Z. Liu	SGCP
	Ms. L. Langlois	PESS
	Mr. R. Steur	NENP
	Mr. M. Khorochev	NENP

CONFERENCE SECRETARIAT

Scientific Secretaries:	J. Kupitz, NENP
	K. Fukuda, NEFW
	V.M.R. Koorapaty, NENP
Conference Service:	K. Morrison
Conference Support:	C. Baghdoyan
	A. Kellner-Windhager
	R. Knoedler
IT Support:	A. Badulescu

LIST OF PANELLISTS

SESSION 1: PANEL ON CONFERENCE THEME

SI No.	Name & address
1	L. Echavarri Director General, OECD/NEA Le Seine-Saint Germain 12, Boulevard des Iles 92130 Issy-Les Moulineaux FRANCE
2	John B. Ritch III, Director General World Nuclear Association 12 th. Floor, Bowater House West 114 Knightsbridge London SW1X 7LJ UNITED KINGDOM
3	Stephen J. Saunders President, Electric Utility Cost Group Inc. 1101 Market Street SP6D Chattanooga, Tennessee 37402-6307, USA
4	L. Tocheny International Science and Technology Center Luganskaya ulitsa 9 P.O. Box 25 115516 Moscow RUSSIAN FEDERATION
5	Evgeny P. Velikhov, President, Academician, Kurchatov Institute (KIAE), Kurchatov Sq. 123181 Moscow Russian Federation

SESSION 4: PANEL ON CHALLENGES FOR THE DEPLOYMENT OF INNOVATIVE TECHNOLOGIES

SI No.	Name & address
1	Roberto Cirimello Senior Advisor, National Atomic Energy Commission Av. Bustillo km. 9.500, (8400) S.C. de Bariloche Rio Negro, Argentina
2	John B. Ritch III Director General, World Nuclear Association 12 th. Floor, Bowater House West, 114 Knightsbridge, London SW1X 7LJ UNITED KINGDOM.
3	Antônio Carlos Barroso de Oliveira Acting President, Comissão Nacional de Energia Nuclear, Rua General Severiano, 90 - Botafogo Rio de Janeiro - RJ – Brazil CEP 22290-901
4	R. Duffey Atomic Energy of Canada Limited (AECL) Chalk River Laboratories Chalk River, Ontario K0J 1JO CANADA
5	Y. A. Sokolov, Ministry of the Russian Federation on Atomic Energy B. Ordynka Street, 24/26, 109180 Moscow RUSSIAN FED
6	J. Vergara Comision Chilena de Energia Nuclear Amunategui No. 95, Casilla 188-D Santiago, Región Metropolitana CHILE
7	Patrice Bernard CEA, Saclay/Batiment 121, DEN/DDIN 91191 Gif-Sur-Yvette Cedex, FRANCE

SI No.	Name & address
8	Ratan K. Sinha, Associate Director, RD&D Group (RED) Bhabha Atomic Research Centre (BARC), Trombay, Mumbai 400 085 INDIA
9	D. Nicholls P.O. Box 9396, Lake Buena Vista Building 1267 Gordon Hood Avenue, Centurion, 0046, SOUTH AFRICA

SESSION 7: CONCLUDING PANEL

SI No.	Name & address
1	R. Chidambaram Chief Scientific Adviser to the Government of India Homi Bhabha Chair Professor, Central Complex, BARC, Mumbai – 400 085, INDIA
2	Evgeny P. Velikhov President, Academician, Kurchatov Institute (KIAE), Kurchatov Sq. 123181 Moscow RUSSIAN FEDERATION
3	R. Duffey Atomic Energy of Canada Limited (AECL) Chalk River Laboratories Chalk River, Ontario K0J 1JO CANADA
4	Björn Wahlström Research Professor, VTT Automation, Systems Engineering, P.O. Box 1301, 02044 VTT, FINLAND
5	Patrice Bernard CEA, Saclay/Batiment 121, DEN/DDIN 91191 Gif-Sur-Yvette Cedex, FRANCE

SI No.	Name & address
6	Shinzou SAITO President, JAERI Tokai-Mura, Naka-gun Ibaraki-Ken 319-1195 JAPAN
7	M. Lawrence Associate Laboratory Director, Pacific Northwest National Laboratory, Mail Stop K8-02 902 Battelle Boulevard Richland, WA 99352 U. S. A.
8	P. C. Florido National Atomic Energy Commission Avenida del Libertador 8250 1429 Buenos Aires ARGENTINA
9	V. M. Mourogov Deputy Director General, Head of the Department of Nuclear Energy Wagramerstrasse 5, A 1400 Vienna, IAEA

LIST OF PARTICIPANTS

Abdallah, A.M.A.	Reactors Department Nuclear Research Center Atomic Energy Authority Postal Code 13759 Cairo EGYPT
Adamov, E.	Research and Development Institute of Power Engineering NIKIET P.O. Box 788 101000 Moscow RUSSIAN FED. Fax: +70959752019 E-mail: slp@entek.ru
Akiyama, T.	Office of International Affairs Japan Atomic Energy Research Institute Baumannstrasse 4/2/13 A-1030 Vienna AUSTRIA Fax: +4317130753 E-mail: akiyama@hems.jaeri.at
Allen, T.	Argonne National Laboratory P.O. Box 2528 Idaho Falls ID 83403-2528 USA Fax: ++208 533 7799 E-mail: todd.allen@anl.gov
Andrews, T.J.	Permanent Mission of the United Kingdom of Great Britain and Northern Ireland to the International Atomic Energy Agency Jaurèsgasse 12 A-1030 Vienna AUSTRIA Fax: +431716134237 E-mail: tim.andrews@fco.gov.uk
Apse, V.A.	Moscow Engineering Physics Institute (MEPhI) MSIEM Kashirskoe Shosse, 31 115409 Moscow RUSSIAN FED. Fax: +70953247026

Astapov, S.V.	Ministry of the Russian Federation on Atomic Energy Staromonetny Pereulok, 26 109180 Moscow RUSSIAN FED. Fax: +70952302420 E-mail: astapov@dies.skc.ru
Atissiouk, V.	Research Laboratory for Nuclear Reactors Tokyo Institute of Technology 2-12-1, O-okayama, Meguro-ku Tokyo 152-8550 JAPAN Fax: ++813 57342959 E-mail: artusyuk@nr.titech.ac.jp
Baldev, R.	Materials, Chemical and Reprocessing Groups Indira Gandhi Centre for Atomic Research Kalpakkam, Tamil Nadu 603 102 INDIA Fax: +914114280301/280060 E-mail: dmg@igcar.ernet.in
Bari, R.A.	Brookhaven National Laboratory P.O. Box 5000 Building 475B Upton NY 11973-5000 USA Fax: +16313441430 E-mail: bari@bnl.gov
Barré, B.	Commissariat à l'énergie atomique (CEA) 29-33 rue de la Fédération F-75752 Paris Cedex 15 FRANCE
Baumann, M.	Technische Universität Graz Institute für Theoretische Physik Petersgasse 16 A-8010 Graz AUSTRIA Fax: +43-316-8738678 E-mail: baumann@itp.tu-graz.ac.at

Baptista, B.D.	IPEN-CNEN Cidade Universitaria Av. Prof. Lineu Prestes 2242 CEP-05508 000 São Paulo, SP BRAZIL Fax: +551138169428 E-mail: bdbfilho@net.ipen.br
Barroso, A.C.O.	Comissão Nacional de Energia Nuclear (CNEN) Rua General Severiano 90, Botafogo BR-22294-900 Rio de Janeiro, RJ BRAZIL Fax: +551138146909 E-mail: jdieguez@ipen.br
Baumann, M.	Technische Universität Graz Institute für Theoretische Physik Petersgasse 16 A-8010 Graz AUSTRIA Fax: +43-316-8738678 E-mail: baumann@itp.tu-graz.ac.at
Bernard, P.	Commissariat à l'énergie atomique (CEA) Saclay Bât.121 DEN/DDIN F-91191 Gif-Sur-Yvette Cedex FRANCE Fax: +33169085891 E-mail: patrice.bernard@cea.fr
Birkhofer, A.	T.U. Munich GERMANY
Bolme, A.	Turkish Atomic Energy Authority Eskisehir Yolu 9 Km. Lodumlu 06530 Ankara TURKEY Fax: +903122878761 E-mail: anil.bolme@taek.gov.tr
Bolshov, L.A.	Nuclear Safety Institute Russian Academy of Sciences (IBRAE) Bolshaya Tulskaya Street, 52 113191 Moscow RUSSIAN FED. Fax: +70959580040 E-mail: bolshov@ibrae.ac.ru

Bonne, A.	Division of Nuclear Fuel Cycle and Waste Technology Department of Nuclear Energy International Atomic Energy Agency P.O. Box 100, A-1400 Vienna AUSTRIA Fax: +43126007 E-mail: a.bonne@iaea.org
Burkart, A.	US Department of State 2201 C. Street N.W. Washington, DC 20520 USA Fax: +1202 647 0775 E-mail: burkartar@t.state.gov
Cantrell, C.L.	Wolf Creek NPP P.O. Box 411 Burlington, Kansas 66831 USA E-mail: chcantr@WCNOC.com
Carre, F.O.	Commissariat à l'énergie atomique (CEA) CEA/DEN/DDIN Saclay Bât. 121 F-91191 Gif-sur-Yvette FRANCE Fax: +33169085892 E-mail: franck.carre@cea.fr
Cavlina, N.	Faculty of Electrical Engineering & Computer University of Zagreb Unska 3 HR-10000 Zagreb CROATIA Fax: +389516129890 E-mail: nikola.cavlina@fer.hr
Celinski, Z.	Institute of Power Engineering and High Voltage Technology Warsaw University of Technology 75, Koszykowa Street PL-00-661 Warsaw POLAND Fax: +4826251914 E-mail: zdceli@ien.pw.edu.pl

Chang, In Soon
Korea Atomic Energy Research Institute
P.O. Box 105
Yuseong
Taejon 305-600
KOREA REP.
Fax: +42 8619611
E-mail: ischang@kaeri.re.kr

Chernilin, Y.F.
Kurchatov Institute
Kurchatov sq. 1
123182 Moscow
RUSSIAN FED.
Fax: +70951963708
E-mail: stukalov@dhtp.kiae.ru

Chidambaram, R.
Bhabha Atomic Research Centre (BARC)
Central Complex
Mumbai 400 085
INDIA
Fax: +91225505151
E-mail: rajachid@rediffmail.com

Cirimello, R.
Bariloche Atomic Centre
National Atomic Energy
Commission
Avenida Bustillo Km. 9.500
Rio Negro
ARGENTINA
Fax: +542944445298
E-mail: cirri@cab.cnea.gov.ar

Clarke, W.L.
Canadian Nuclear Association
130 Albert Street, Suite 1610
Ottawa, Ontario K1P 5G4
CANADA
Fax: +16132370989
E-mail: clarkew@cna.ca

Corak, Z.
Institute for Nuclear Technology
INETEC
Koturaska 51
HR-10000 Zagreb
CROATIA
Fax: +385 1 6530849
E-mail: zrinka.corak@inetec.hr

Darilek, P.
VUJE Trnava Inc.
Engineering Design and Research Organisation
Okruzna 5
SK-91864 Trnava
SLOVAKIA
E-mail: darilek@vuje.sk

Degueldre, C.
Paul Scherrer Institute
CH-5232 Villigen PSI
SWITZERLAND
E-mail: claude.degueldre@psi.ch

Delmastro, D.
National Atomic Energy Commission
Avenida del Libertador 8250
1429 Buenos Aires
ARGENTINA
Fax: +54294445299
E-mail: delmast@cab.cnea.gov.ar

DeMella, J.R.
Nuclear Power Economics
120 Millstone Road
Glastonbury CT 06033
USA
Fax: +8606331506
E-mail: demeljr@aol.com

Depisch, F.
Division of Nuclear Power
Department of Nuclear Energy
International Atomic Energy Agency
P.O. Box 100 A-1400 Vienna
AUSTRIA
Fax: +43126007
E-mail: f.depisch@iaea.org

Dmitriev, A.
Federal Nuclear and Radiation
Safety Authority of Russia
(RF Gosatomnadzor)
Taganskaya Street 34
109147 Moscow
RUSSIAN FED.
Fax: +70959124041
E-mail: ulyanoralas@gan.ru

Duffey, R.B.
Atomic Energy of Canada Limited (AECL)
Chalk River Laboratories
Chalk River, Ontario K0J 1JO
CANADA
E-mail: duffeyr@aecl.ca

Dumitrache, I. — Institute for Nuclear Research Pitesti
Strada Campului No. 1, Mioveni
P.O. Box 78
R-0300-Pitesti, Jud. Arges
ROMANIA
Fax: +40248262449
E-mail: iondum@pitesti.ro

Dzhalavyan, A.V. — Research and Development
Institute of Power Engineering
NIKIET
P.O. Box 788
101000 Moscow
RUSSIAN FED.
Fax: +70959752019
E-mail: avd@entek.ru

Echávarri, L. — OECD/NEA
Le Seine St-Germain
12, Boulevard des Iles
F-92130 Issy-les-Moulineaux
FRANCE
Fax: +33145241110
E-mail: luis.echavarri@oecd.org

El Sharaky, M.A.M. — Nuclear Power Plants Authority
P.O. Box 108
Code No. 113811
Abbassia
Nasr City, Cairo 1371
EGYPT
Fax: +2022616476
E-mail: nppa1@idsc.net.eg.

El Shehawy, I.A. — Nuclear Power Plants Authority
P.O. Box 108
Code No. 113811
Abbassia
Nasr City, Cairo 1371
EGYPT
Fax: +2022616476
E-mail: nppa1@idsc.net.eg

Eltawila, F. — Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory
Commission
Mail stop T-10 F12
Washington, DC 20555
USA

Emelyanov, V.S.
Ministry of the Russian
Federation on Atomic Energy
Staromonetny Pereulok, 26
109180 Moscow
RUSSIAN FED.
Fax: +70959533053
E-mail: emelyanov@dae.minatom.ru

Feretic, D.
University of Zagreb
c/o Faculty of Electrical
Engineering and Computing
Unska 3
HR-41000 Zagreb
CROATIA
Fax: +38516129890
E-mail: danillo.feretic@fer.hr

Fiorini, G.L.
Commissariat à l'énergie atomique (CEA)
CEA/DEN/DDIN
Saclay Bât. 121
F-91191 Gif-sur-Yvette
FRANCE
Fax: +33169085892
E-mail: gian-luigi.fiorini@cea.fr

Florido, P.C.
National Atomic Energy Commission
Avenida del Libertador 8250
1429 Buenos Aires
ARGENTINA
Fax: +54294445178
E-mail: florido@cab.cnea.edu.ar

Fukuda, K.
Division of Nuclear Fuel
Cycle and Waste Technology
Department of Nuclear Energy
International Atomic Energy Agency
P.O.Box 100, A-1400 Vienna
AUSTRIA
Fax: +43126007
E-mail: k.fukuda@iaea.org

Fujjie, Y.
Atomic Energy Commission (AEC)
Ministry of Education, Culture, Sports, Science & Technology
3-2-2 Kasumigaseki, Chiyoda-Ku
Tokyo 100-0013
JAPAN
Fax: +8133589149

Gabaraev, B.A.	Research and Development Institute of Power Engineering P.O. Box 788 101000 Moscow RUSSIAN FED. Fax: +70959752019 E-mail: gabaraev@entek.ru
Gagarinskaya, I.	Russian Research Centre Kurchatov Institute Kurchatov Square, 1 123182 Moscow RUSSIAN FED. Fax: +70951962073 E-mail: agagarin@kiae.ru
Gagarinski, A.	European Nuclear Society c/o Kurchatov Institute Kurchatov Square, 1 123182 Moscow RUSSIAN FED. Fax: +7957976014 E-mail: agagarin@kiae.ru
Ganguly, C.	Government of India Department of Atomic Energy Nuclear Fuel Complex Chairman & Chief Executive E.C.I.L. (P.O) Hyderabad 500 062 INDIA Fax: +91407121305 E-mail: ganguly@nfc.ernet.in
Gauthier, J-C.	Commissariat à l'énergie atomique (CEA) CEA/DEN/DDIN Saclay Bât. 121 F-91191 Gif-sur-Yvette FRANCE Fax: +33169088148 E-mail: jean-claude.gauthier@cea-fr
Gazetov, R.	Intellectual Property Development Fund (IPDF) Jaunystes g. I-20 LT-4761 Visaginas LITHUANIA Fax: +37038671674 E-mail: r.gazetov@ipdf.lt; lumograf@tts.lt

Gevorgyan, A.A.
Department of Atomic Energy
Ministry of Energy
Government House
2 Republic Square
Yerevan 375010
ARMENIA
Fax: +3741562776
E-mail: atomen@freenet.am

Gill, C.
Ministry of Economic Affairs
North Gate III
Boulevard du Roi Albert II, 16
B-1000 Bruxelles
BELGIUM
Fax: +3222065711
E-mail: carole.gill@mineco.fgov.be

Glebov, V.B.
Moscow Engineering Physics Institute (MEPHI)
MSIEM
Kashirskoe Shosse, 31
115409 Moscow

Gomez de Soler, S.
National Atomic Energy Commission
Avenida del Libertador 8250
1429 Buenos Aires
ARGENTINA
Fax: +67727421/7833
E-mail: gomsoler@cnea.gov.ar;bronte@cnea.gov.ar

Greneche, D.
COGEMA
2, rue Paul Dautier
BP 4
F-78141 Velizy Cedex
FRANCE
Fax: 0033139485164
E-mail: dgrenèche@cogema.fr

Gudowski, W.
Nuclear and Reactor Physics
Royal Institute of technology
Alba Nova University Centre
S-10691 Stockholm
SWEDEN
Fax: +46855378465
E-mail: wacek@neutron.kth.se

Guindon, S.	Nuclear Energy Division Natural Resources Canada 580 Booth Street Ottawa, Ontario K1A 0E4 CANADA Fax: +16139950087 E-mail: sguindon@nrcan.gc.ca
Hamid, A.	Pakistan Atomic Energy Commission P.O. Box 1114 Islamabad PAKISTAN Fax: +92519204908
Hauser, B.	International Institute for Applied Systems Analysis Schlossplatz 1 A-2361 Laxenburg AUSTRIA Fax: +43223671313 E-mail: hauser@iiasa.ac.at
Heindler, M.	Technische Universität Graz Institute für Theoretische Physik Petersgasse 16 A-8010 Graz AUSTRIA Fax: +43-316-8738678 E-mail: heindler@itp.tu-graz.ac.at
Henzel, V.	Nuclear Research Center - Negev Israel Atomic Energy Commission P.O. Box 9001 Beer-Sheva 84190 ISRAEL Fax: +97286567199 E-mail: uhenzel@netvision.net.il
Herrera, L.	Nuclear Energy Research Initiative (NERI) Office of Advanced Nuclear Research NE-20 Germantown Building 1000 Independence Ave, SW Washington, DC 20585 USA Fax: +13019035057

Hiraiwa, K. | Nuclear System Group
System Design and Engineering Department
Toshiba Corp.
8 Shinsugita-cho, Isogo-ku
Yokohama 235-8523
JAPAN
Fax: +81457702179
E-mail: kouji.hiraiwa@toshiba.co.jp

Hittner, D. | MICANET
Tour Framatome
Cedex 16
F-921084 Paris-La-Défense
FRANCE
Fax: +33147961509
E-mail: dhittner@framatome.fr

Hopwood, J. | Atomic Energy of Canada Limited (AECL)
2251 Speakman Drive
Mississauga
Ontario, LSK 1B2
CANADA
Fax: +19054037337
E-mail: hopwoodj@aecl.ca

Hugon, M. | European Commission
MO75 5155
B-1049 Bruxelles
BELGIUM
Fax: +3222954991
E-mail: michel.hugon@cec.eu.int

Iizuka, K. | Japan Atomic Energy Research Institute (JAERI)
14-1 Seuhiro-cho, Kashiwa-shi
Chiba 277-0842
JAPAN
Fax: +81471422372
E-mail: sachiko@hems.jaeri.go.jp

Jelinek-Fink, P. | Mittelgasse 27/9, A
A-1060 Vienna
AUSTRIA
E-mail: p.jelinek@vienna.at

Jenkins, P.R.	Ambassador Permanent Mission of the United Kingdom of Great Britain and Northern Ireland to the IAEA Jaurèsgasse 12 A-1030 Vienna AUSTRIA Fax: +431716134900 E-mail: peter.jenkins@fco.gov.uk
Jorant, C.	International Department - AREVA 27-29, rue Le Peletier F-75433 Paris Cedex 09 FRANCE E-mail: caroline.jorant@arevagroup.com
Juhn, P.E.	Division of Nuclear Power Department of Nuclear Energy International Atomic Energy Agency P.O. Box 100 A-1400 Vienna AUSTRIA Fax: +4312600 7 22750 E-mail: p.e.juhn@iaea.org
Kakodkar, A.	Atomic Energy Commission Anushakti Bhavan Chhatrapati Shivaji Maharaj Marg Mumbai 400 001 INDIA Fax: +91 222843888 E-mail: kakodkar@magnum.barc.ernet.in
Katanishi, S.	Japan Atomic Energy Research Institute (JAERI) Oarai-machi Ibaraki-ken 311-1394 JAPAN Fax: +81292648710 E-mail: katanishi@spa.oarai.jaeri.go.jp
Kataoka, K.	Nuclear System Group System Design and Engineering Department Toshiba Corp. 8 Shinsugita-cho, Isogo-ku 235-8523 Yokohama JAPAN Fax: +814577021792298 E-mail: kazuyoshi.kataoka@toshiba.co.jp

Kim, Yong-hwan	Permanent Mission of the Republic of Korea to the International Atomic Energy Agency Gregor Mendel Strasse 25 A-1180 Vienna AUSTRIA Fax: +4314781013 E-mail: yhk113@yahoo.co.kr
Klimenko, A.	Elektrostal P.O. Box 9 Moscow Region 144009 RUSSIAN FED. Fax: ++7 09657 70546 E-mail: isez@enezgya.ru
Knoglinger, E.	Am Winklerwald 15 A-4020 Linz AUSTRIA E-mail: e.knoglinger@utanet.at
Kodochigov, N.G.	Experimental Bureau of Mechanical Engineering OKBM Burnakovsky proezd, 15 603603 Nizhny Novgorod RUSSIAN FED. Fax: +8312418772 E-mail: icdokbm@nnov.city.ru
Kosovan, S.	Hydrogen Institute Vajnorsa 58/9 Bratislava SLOVAKIA Fax: ++421244633546
Koorapaty, V.M.R.	Division of Nuclear Power Department of Nuclear Energy International Atomic Energy Agency P.O. Box 100 A-1400 Vienna AUSTRIA Fax: ++43126007 E-mail: v.m.r.koorapaty@iaea.org
Korinny, A.O.	Khmelnitsky Nuclear Power Plant appzt 123, 10 Mira str. Neteshin 30100 Khmelnitsky Region UKRAINE Fax: +442944875

Koyama, T. | Central Research Institute of Electric Power Industry
2-11-1 Iwadokita, Komae-shi
Tokyo 201-8511
JAPAN
Fax: +81334807956
E-mail: koyama@criepi.denken.or.jp

Kozel, L.V. | Ministry of the Russian Federation on Atomic Energy
Staromonetny Pereulok, 26
109180 Moscow
RUSSIAN FED.
Fax: +70952302420
E-mail: astapov@dies.skc.ru

Královec, J. | Nuclear Power and Safety Division
Nuclear Research Institute Rez
CZ-250 68 Rez
CZECH REP.
Fax: +420220941029
E-mail: kjo@ujv.cz

Kryuchkov, E.F. | Moscow Engineering Physical Institute (MEPHI)
MSIEM
Kashirskoe Shosse, 31
115409 Moscow
RUSSIAN FED.
E-mail: efk@nr.mephi.ru

Kupitz, J. | Division of Nuclear Power
Department of Nuclear Energy
International Atomic Energy Agency
P.O. Box 100
A-1400 Vienna
AUSTRIA
Fax: ++431 26007
E-mail: j.kupitz@iaea.org

Kuriyama, M. | Office of International Affairs
Japan Atomic Energy
Research Institute
Baumannstrasse 4/2/13
A-1030 Vienna
AUSTRIA
Fax: +4317130753
E-mail: kuriyama@jaeri.at

Kusuno, S. Institute of Applied Energy
Shinbashi SY Bldg.
14-2 Nishishinbashi 1-chome
Minato-ku
Tokyo 105-0003
JAPAN
Fax: +81335011735
E-mail: kusuno@iae.or.jp

Kuznetsov, V. Division of Nuclear Power
Department of Nuclear Energy
International Atomic Energy Agency
P.O. Box 100
A-1400 Vienna
Fax: +4312600 7 29598
E-mail: v.v.kuznetsov@iaea.org

Larsson, S.E. Carl Bro Energikonsult AB.
S-20509 Nalmö
SWEDEN
Fax: +4640305916
E-mail: stig-erik.larsson@calbro.se

Lasman, A.N. P2SRM BATAN
Center for Development of Advanced Reactor System
NNEAI
Mampang Prapatan
Jakarta 12710
INDONESIA
Fax: +622175872029
E-mail: asnatio@batan.go.id;asnatio@yahoo.com

Lavrinovich, A.A Federal Nuclear and Radiation
Safety Authority of the Russian Federation
(RF Gosatomnadzor)
Taganskaya Street, 34
109147 Moscow
RUSSIAN FED.
Fax: +70959124041
E-mail: laa@hq.gan.ru

Lawrence, M. Pacific Northwest National Laboratory
Mail Stop K8-02
902 Battelle Boulevard
Richland, WA 99352
USA
Fax: ++509372 4774

LeCoyote, R. Greenpeace International

Lebadezet, M.	FRAMATOME ANP Tour Areva 1, place de la Coupole F-92084 Paris-La-Défense Cedex FRANCE Fax: +33147961509 E-mail: marcel.lebadezet@framatome-anp.com
Leiser, H.K.	US Department of Energy 1000 Independence Ave, SW Washington, DC 20585-1290 USA Fax: +12025868353 E-mail: helen.leiser@hq.doe.gov
von Lensa, W.	Institute for Safety Research and Reactor Technology Forschungszentrum Jülich Leo-Brandt-Strasse Building 09.11 D-52425 Jülich GERMANY Fax: +492461615342 E-mail: w.von.lensa@fz-juelich.de
Litvinsky, L.	State Scientific and Research Centre of the Control and Emergency Response Systems Geroev Stalingrada str., 64/56 04213 Kiev UKRAINE Fax: +380444119270 E-mail: litvinsky@dnic.kiev.ua
Lopatkin, A.V.	Research and Development Institute of Power Engineering NIKIET P.O. Box 788 101000 Moscow RUSSIAN FED. Fax: +70959752019 E-mail: lopatkin@entek.ru
Loriette, P.	Électricité de France 1, avenue du Général de Gaulle F-92141 Clarmart Cedex FRANCE Fax: +33147653499 E-mail: philippe.loriette@edf.fr

Maliverney, B.	Électricité de France 1, avenue du Général de Gaulle F-92141 Clarmart Cedex FRANCE Fax: +33147655173 E-mail: bernard.maliverney@edf.fr
Minato, A.	Central Research Institute of Electric Power Industry (CRIEPI) 2-11-1 Iwado-kita, Komae-shi Tokyo 211-8511 JAPAN E-mail: minato@criepi.denken.or.jp
Murray, J.	World Energy Council 5th Floor, Regency House 1-4 Warwick Street London W1B 5LT UK Fax: +442077345926 E-mail: murray@worldenergy.org
Mourogov, V.M.	Department of Nuclear Energy International Atomic Energy Agency P.O. Box 100 A-1400 Vienna AUSTRIA Fax: +43126007 E-mail: v.mourogov@iaea.org
Muzikova, E.	Institute for Safety Research and Reactor Technology Forschungszentrum Jülich Leo-Brandt-Strasse Building 09.11 D-52425 Jülich GERMANY Fax: +492461615242 E-mail: e.muzikova@fz-juelich.de
Nakagami, Y.	Japan Nuclear Cycle Development Institute (JNC) 4-49 Muramatsu Tokai-mura, Naka-gun Ibaraki-ken 319-1184 JAPAN Fax: +81292824904 E-mail: nakagami@hq.jnc.go.jp

Nicholls, D.R.
PMBR (Pty) Ltd.
Lake Buena Vista Building
1267 Gordon Hood Avenue
P.O. Box 9396
Centurion 0046
SOUTH AFRICA
Fax: +27126779433
E-mail: david.nicholls@pmbr.co.za

Novikov, V.M.
International Institute of Applied Systems Analysis
Schlossplatz 1
A-2361 Laxenburg
AUSTRIA
Fax: +431223671313
E-mail: novikov@iiasa.ac

Okubo, T.
Japan Atomic Energy Research Institute (JAERI)
2-4 Shirakata Sirone
Tokai-mura, Naka-gun
Ibaraki-ken 319-1195
JAPAN
Fax: +81292825276
E-mail: okubo@hens.jaeri.go.jp

Oussonov, V.I.
State Scientific Centre
Institute of Physics and Power
Engineering (IPPE)
Bondarenko Sq. 1
249033 Obninsk, Kaluga Region
RUSSIAN FED.
Fax: +70843998413
E-mail: vouss@ippe.obninsk.ru

de Oliveira Barroso, A.C.
Commissão Nacional de Energia Nuclear (CNEN)
Rua general Severiano 90
Botafogo
22 294-900 Rio de Janeiro, R.J.
BRAZIL

Polunichev, V.I.
Experimental Bureau of Mechanical Engineering
OKBM
Burnakovsky Proezd, 15
603074 Nizhny Novgorod
RUSSIAN FED.
Fax: +78312418772
E-mail: icdokbm@nnov.city.ru

Poplavsky, V.M.
State Scientific Centre
Institute of Physics and Power Engineering (IPPE)
Bondarenko sq., 1
249033 Obninsk, Kaluga Region
RUSSIAN FED.
Fax: +70952302326
E-mail: vmpopl@obninsk.ru

Rachkov, V.
Ministry of the Russian Federation on Atomic Energy
B. Ordynka Street, 24/26
109180 Moscow
RUSSIAN FED.
Fax: +70959517693
E-mail: rachkov@minatom.net

Rashad, S.M.
Nuclear Center for Nuclear Safety and Radiation Control
3 Ahmed El Zomar
Nasr City, Cairo 7551
EGYPT
Fax: +20227402378;+2022876031

Rauf, T.
Office of External Relations and Policy Co-ordination
International Atomic Energy Agency
P.O. Box 100
A-1400 Vienna
AUSTRIA
Fax: +43126007
E-mail: t.rauf@iaea.org

Ritch III, J.B.
World Nuclear Association
Bowater House West, 12th Floor
114 Knightsbridge
London SW1X 7LJ
UK
Fax: +442072250308
E-mail: ritch@world-nuclear.org

Rogner, H.H.
Department of Nuclear Energy
International Atomic Energy Agency
P.O. Box 100
A-1400 Vienna
AUSTRIA
Fax: +431 26007
E-mail: H.H.Rogner@iaea.org

Romenkova, O.
Federal Nuclear and Radiation
Safety Authority of Russia
(RF Gosatomnadzor)
Taganskaya Street, 34
109147 Moscow
RUSSIAN FED.
Fax: +70959124041

Sagayama, Y.
FBR Cycle System Development Office
Japan Nuclear Cycle Development Institute
4002 Narita-cho, O-arai-machi
Higashi, Ibaraki-gun 311-1393
JAPAN
Fax: +81292677173
E-mail: sagayama@oec.jnc.go.jp

Saito, M.
Research Laboratory for Nuclear Reactors
Tokyo Institute of Technology
2-12-1, O-okayama, Meguro-ku
Tokyo 152-8550
JAPAN
Fax: +81357342959
E-mail: msaitoh@nr.titech.ac.jp

Saito, Shinzou
Japan Atomic Energy Research Institute (JAERI)
14-1 Seuhiro-cho, Kashiwa-shi
Chiba 277-0842
JAPAN
Fax: +81471422372
E-mail: sachiko@hems.jaeri.go.jp

Samdani, G.
Pakistan Atomic Energy Commission
P.O. Box 1114
Islamabad
PAKISTAN
Fax: +92519204908
E-mail: samdani@dgcc.org.pk

Samejima, K.
Central Research Institute of Electric Power Industry
Nuclear Energy Systems Department
1-6-1, Ohtemachi, Chiyoda-ku
Tokyo 100-8126
JAPAN
Fax: +81332872840
E-mail: samejima@criepi.denken.or.jp

Sato, K.	Japan Nuclear Cycle Development Institute 4002, Narita-cho, Oarai-machi Higashi-Ibaraki-gun Ibaraki-ken 311-1393 JAPAN Fax: +81292677173 E-mail: ksat@hq.jnc.go.jp
Saunders, S. J.	Electric Utility Cost Group Inc. Tennessee Valley Authority 1101 Market Street SP6D Chattanooga Tennessee 37402-6307 USA Fax: +1 4237516307 E-mail: sjsaunders@tva.gov
Sebian, V.S.	Department of Nuclear Physics and Technology Slovak University of Technology Ilkovicova 3 SK-812 19 Bratislava SLOVAKIA Fax: +421265327207 E-mail: sebianv@hotmail.com
Sefidvash, F.	Nuclear Engineering Dept. Federal University of Rio Grande do Sul Av. Osvaldo Aranha 99 4 Andar 90046-900 Porto Alegre, RS BRAZIL Fax: +555133163983 E-mail: farhang@ufrgs.br
Shmelev, A.N.	Moscow Engineering Physics Institute (MEPHI) Kashirskoe Shosse, 31 115409 Moscow RUSSIAN FED. E-mail: shmelev@nr.mephi.ru
Sinha, R.K.	Bhabha Atomic Research Centre (BARC) Reactor Design & Development Mumbai 400 085 INDIA Fax: +91225505303 E-mail: rksinha@magnum.barc.ernet.in;sinhark@vsnl.com

Sokolov, Y.A.
Ministry of the Russian Federation on Atomic Energy
B. Ordynka Street, 24/26
109180 Moscow
RUSSIAN FED.
Fax: +70959517693
E-mail: sokolov@minatom.ru

Subbotin, S.A.
Kurchatov Institute
Kurchatov sq., 1
123182 Moscow
RUSSIAN FED.
Fax: +70951963708
E-mail: subbotin@dhtp.kiae.ru

Suzuki, T.
Central Research Institute of Electric Power Industry (CRIEPI)
1-6-1 Ohetamchi, Chiyoda-ku
Tokyo 100-8126
JAPAN
Fax: +81332872805
E-mail: tatsu@criepi.denken.or.jp

Teller, A.T.
FORATOM
Rue Belliard 15-17
B-1040 Bruxelles
BELGIUM
Fax: +3225023902
E-mail: andrew.teller@foratom.org

Tocheny, L.
International Science and Technology Centre
Luganskaya ulitsa 9
P.O. Box 25
115516 Moscow
RUSSIAN FED.
Fax: +70957976028
E-mail: tocheny@ista.ru

Torres, G.
Comision Chilena de Energia Nuclear
Amunategui No. 95
Casilla 188-D
Santiago, Región Metropolitana
CHILE
Fax: +5626990735
E-mail: gtorres@vtr.net

Toshinsky, G.I.
Institute for Physics and Power Engineering
Bondarenko Square, 1
249033 Obninsk, Kaluga Region
RUSSIAN FED.
E-mail: toshinsky@ippe.obninsk.ru

Trinh Dinh, Hai
Dalat Nuclear Research Institute
1 Nguyen Tu Lac
Dalat
VIETNAM
E-mail: trdhai@hem.vnn.vn

Tsuchie, Y.
Japan Atomic Power Company
Mitoshiro Bldg.
1-1 Kanda-Mitoshiro-cho,
Chiyoda-ku
Tokyo 101-0053
JAPAN
Fax: +81344156690
E-mail: yasuo-tsuchie@japc.co.jp

Tsuji, H.
Japan Atomic Energy Research Institute (JAERI)
14-1 Seuhiro-cho, Kashiwa-shi
Chiba 277-0842
JAPAN
Fax: +81471422372
E-mail: sachiko@hems.jaeri.go.jp

Turkenburg, W.C.
Department of Science
Technology and Society
Utrecht University
Room B219 Centrumgebouw Noord
Padualaan 14
NL-3584 CH Utrecht
NETHERLANDS
Fax: +31302537601
E-mail: w.c.turkenburg@chem.uu.nl

Peucker, J.
Framatome ANP GmbH
Abteilung FGTM
Bundenstr. 43
D-91058 Erlangen
GERMANY
E-mail: joerg.peucker@framatome-anp.com

Popov, D.
Department of Thermal and Nuclear Power Engineering
Technical University of Sofia
8 Kliment Ohridski St. EMF
Sofia-1000
BULGARIA
Fax: +35929652295
E-mail: dpopov@tu-sofia.bg

Roglans-Ribas, J.	Nuclear Engineering Division Argonne National Laboratory 9700 South Cass Avenue Argonne, IL 60439 USA Fax: +16302524500 E-mail: roglans@anl.gov
Romano, R.	CERCA Tour Areva F-92084 Paris-La-Défense FRANCE Fax: +33147965892 E-mail: romanorene@hotmail.com
Velikhov, E.P.	Russian Research Centre Kurchatov Institute (KIAE) Kurchatov Square 123181 Moscow RUSSIAN FED. Fax: +70959430023 E-mail: evp@evp.kiae.ru
Vergara, J.A.	Comision Chilena de Energia Nuclear Amunategui No. 95 Casilla 188-D Santiago, Región Metropolitana CHILE Fax: +5624702600 E-mail: jvergara@alum.mit.edu
Vishnevsky, Yu.	Federal Nuclear and Radiation Safety Authority of Russia (RF Gosatomnadzor) Taganskaya Street, 34 109147 Moscow RUSSIAN FED. Fax: +70959124041 E-mail: las@gan.ru
Wahlström, B.	Technical Research Centre VTT Industrial Systems P.O. Box 1301 FIN-02044 VTT FINLAND Fax: +35894566752 E-mail: bjorn.wahlstrom@vtt.fi

Wakabayashi, J.	Japan Atomic Energy Research Institute (JAERI) 14-1 Seuhiro-cho, Kashiwa-shi Chiba 277-0842 JAPAN Fax: +81471422372 E-mail: sachiko@hems.jaeri.go.jp
Wider, H.U.	Joint Research Centre of the EC Institute for Energy Westerduinweg 3 NL-1755 Le Petten NETHERLANDS Fax: +31224565621 E-mail: wider@jrc.nl
Wójcik, T.	National Atomic Energy Agency 36 Krucza Str. PL-00-921 Warsaw POLAND Fax: +48226959804 E-mail: tad.wojcik@wp.pl
Xerri, C.	International Department AREVA 27-29 rue Le Peletier F-75433 Paris cedex 09 FRANCE Fax: +33139262780 E-mail: christophe.xerri@arevagroup.com
Yang, Myung Seung	Korea Atomic Energy Research Institute Expo Apt 108-1404 Jummin-dong, Yueseong-gu Taejon KOREA REP. E-mail: msyangl@kaeri.re.kr
Yokota, K.	4-25-1-305 Ukima Kita-ku Tokyo JAPAN Fax: ++813 3581 9828 E-mail: kyokota1@op.cao.go.jp
Yoshikawa, T.	Japan Nuclear Cycle Development Institute 4-49 Muramatsu,Tokai-mura Naka-Gun Ibaraki-ken 319-1184 JAPAN Fax: +81292828049 E-mail: yoshika@hq.jnc.go.jp

Zaker, M.	Nuclear Research Centre Atomic Energy Organization of Iran End of Kargar Avenue P.O. Box 11365-8486 Tehran IRAN,ISL.REP Fax: +98218021412 E-mail: mzakar@seai.neda.net.ir
Zverev, K.V.	Ministry of the Russian Federation on Atomic Energy Staromonetny Pereulok, 26 109180 Moscow RUSSIAN FED. Fax: +70959533053 E-mail: zverev@dae.minatom.ru