# Year Report, 2007

# IAEA Research Contract No. 13093

## Concept of Small Power Reactor Installation without Refueling during Lifetime (SVBR-75/100)


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## CONTENTS

## Nomenclature

| | |
|---|---|
| AHRS | – autonomous heat removal system |
| BR | – breeding ratio |
| CBR | – core breeding ratio |
| CPS | – control and protection system |
| EFPD | – effective full power days |
| FP | – fission products |
| FR | – fast reactor |
| HRC | – heat removal condenser |
| LBC | – lead-bismuth coolant |
| LWR | – light water reactor |
| MA | – minor actinides |
| MCC | – main circulating circuit |
| MCP | – main circulation pump |
| MOX fuel | – mixed oxide fuel |
| NFC | – nuclear fuel cycle |
| NP | – nuclear power |
| NPP | – nuclear power plant |
| NS | – nuclear submarine |
| PHRS | – passive heat removal system |
| R&D | – research and development |
| RAW | – radioactive waste |
| RBMK | – large size channel type reactor |
| RI | – reactor installation |
| RMB | – reactor monoblock |
| SG | – steam-generator |
| SNF | – spent nuclear fuel |
| SS | – safety system |
| SVBR | – lead-bismuth fast reactor |
| TR | – thermal reactor |
| TRUOX fuel | – oxide fuel based on mixture of uranium, plutonium and minor actinides (MA) |
| UN | – uranium mono-nitride |
| WWER | – water cooled water moderated power reactor |

In compliance with a working plan for the Third Year of the Contract the results of work on following items have to be presented:

1. Development of the principal hydraulic diagram of the reactor installation (RI).
2. Selection of the primary and secondary circuits' coolant parameters.
3. Fuel cycle concept development.
4. Nonproliferation concept development.

The results of carried out work are presented below:

### 1. Development of the principal hydraulic diagram of the RI.

The principal scheme of the SVBR-75/100 is presented in Fig. 1.

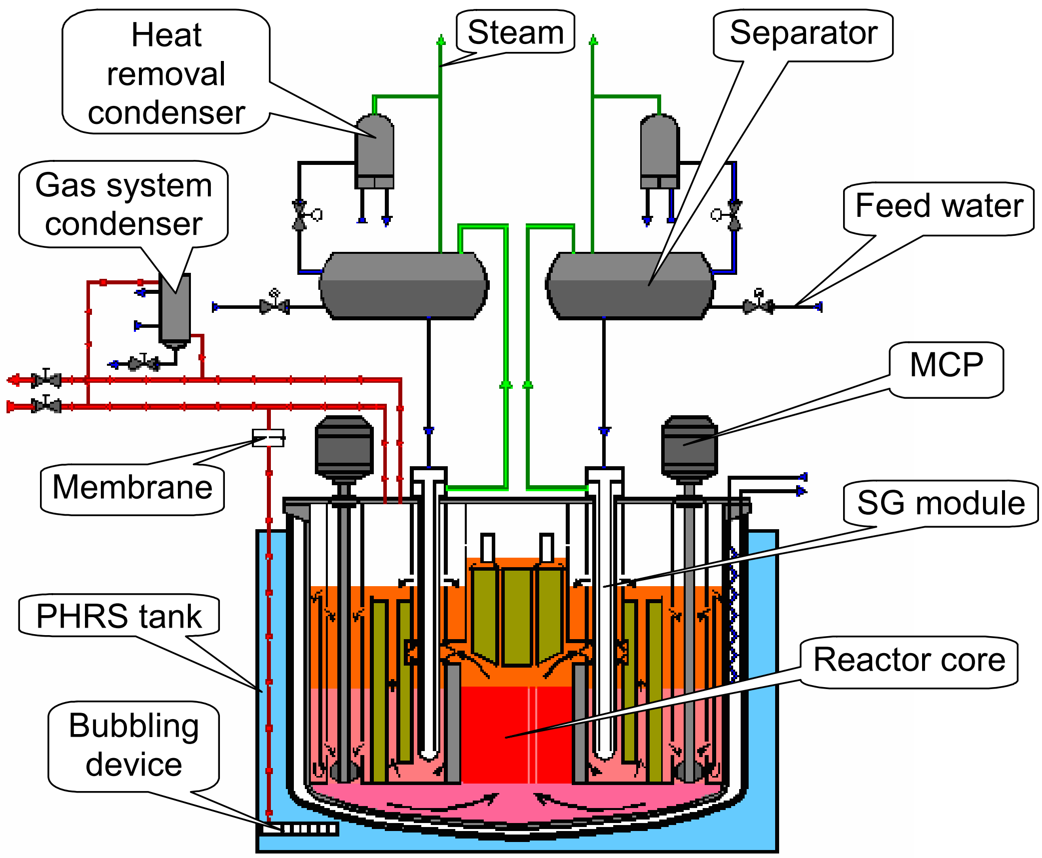


Fig. 1 The principal hydraulic scheme of the SVBR-75/100 RI

The reactor installation includes [1]:

- the reactor monoblock;
- the equipment and pipelines of the primary circuit gas system;
- the equipment and pipelines of the secondary circuit;
- the equipment and pipelines of the passive heat removal system;
- the equipment of the control and diagnostics system;
- the control sub-systems of the reactor installation components;
- the transport and technological equipment of the nuclear fuel management system (for use in developing countries, the equipment of the nuclear fuel management system is not supplied to the user), the equipment for reactor assembly and balance and

commissioning, and the equipment for repair and maintenance; for the modular power units it is acceptable to have a single set of equipment regardless of the number of such modules in the power unit.

### 1.1. Primary circuit circulation scheme

The hydraulic connections over the coolant among the equipment of the primary circuit's system, which form two circuits of coolant's circulation (the main and auxiliary ones), are wholly formed within the reactor monoblock (RMB) vessel by the components of the in-vessel structures without using the pipelines and valves.

Within the main circulation circuit (MCC) the coolant flows according to the following scheme. Being heated in the core, coolant flows to the inlet of the medium part of the inter-tube space of the twelve steam-generators (SG) modules switched on in parallel.

Then coolant is divided into two flows. One flow moves bottom-up in the inter-tube space and is transferred into the peripheral buffer chamber with a free level of the "cold" coolant. Another flow moves top-down and is transferred into the outlet chamber out of which it is transferred to the channels into in-vessel radiation shielding. Then it is transferred into the peripheral buffer chamber as well. Out of the peripheral buffer chamber the main coolant flow is transferred over the downcomer circular channel along the RMB vessel via the inlet chamber to the main circulation pump's (MCP) suction. Another part of coolant is transferred to the MCP suction over the circular channel formed with a MCP vessel and shaft. Out of the MCP the coolant is transferred along the two channels organized in the block of the lower zone of in-vessel radiation shielding into the distributing chamber, from which it is transferred to the reactor inlet chamber thus closing the MCC circuit.

An auxiliary circuit for coolant's circulation is formed in the channels designed for mounting the jackets of the absorbing rods of the reactor's control and protection system (CPS), in the channels in the reactor's shielding plug and is purposed to organize cooling of the CPS absorbing rods, providing the required temperature mode in the central buffer chamber and mass-exchangers' channels for increasing of oxygen concentration in the coolant.

### 1.2. Primary circuit gas system

The gas system is intended for the following:

- maintaining the inert atmosphere above the free surfaces of the primary circuit coolant in the monoblock;
- removing the steam from the primary circuit if the inter-circuit leak of the SG occurs;
- providing the coolant filling and draining in the monoblock;
- monitoring the tightness of the core fuel elements' claddings.

High purity argon is used as an inert gas in the system. The gas system's operation pressure of 0.103 MPa (abs.) is used, which ensures the pressure excess over the atmospheric pressure.

The gas system connects the monoblock's gas spaces, including the central and peripheral buffer chambers, gas spaces of the SG modules, circulation pumps.

The primary circuit gas system performs a function of protection and localizing safety system by removing steam from the primary circuit's cover gas to the gas system's condenser and then to the reservoir for condensate intake in an event of inter-circuit leak in the SG modules. The system of removing steam from the gas system is designed for guillotine rupture of a single tube of the evaporator module. At the same time the gas system is also a technical tool for overcoming an accident being beyond the design basis, which is caused by a postulated guillotine rupture of several tubes of the evaporator module.

When localization of the SG leak is performed, one of the basic elements is a LBC circulation circuit with coolant streams up-going to the free coolant's levels. This provides the reliable separation of steam-water mixture and prevents the drag of steam to the core by a descending primary circuit's coolant stream.

In events of small SG leaks (up to full damage of a single SG tube), two gas system's condensers cooled by technical water are used. At this, their efficiency enables to maintain pressure in the gas system of the RI within 0.5 MPa.

To withstand the large SG leaks (postulated damage of several SG tubes – an accident that is beyond the design basis), the gas system is connected with the passive heat removal system's (PHRS) tank. This line is overlapped by a membrane-protection device designed for the rupture at the gas system pressure of 1 MPa that is not dangerous for the RMB vessel. At this, the volatile radionuclides of the cover gas will remain in the tank water but the radioactive non-condensing gases will escape via a filtrate system to the atmosphere. Radioactivity release will not exceed the permissible level.

### 1.3. Secondary circuit circulation scheme

The secondary circuit system is intended for obtaining saturated steam with the specified parameters for turbine installation by using heat removed to the SG from the primary circuit and for scheduled and emergency cooling of the RI.

The secondary circuit system includes:

- two SGs, each including 6 modules installed in the monoblock, a steam separator with boiler water and steam-water mixture pipelines with shut-off valves and safety valves;
- feeding water pipelines with automatically controlled adjusting and shut-off valves;
- saturated steam pipelines with automatically controlled shut-off valves;
- pipelines of blowing-through the separators with automatically controlled shut-off valves;
- the autonomous heat removal system with condensers and adjusting valves.

### 1.4 Autonomous heat removal system of the reactor

The autonomous heat removal system (AHRS) is a protection safety system designed for decay heat removal from the reactor core. Besides, this system is used for starting up and shutting down the RI and implementing other technological regimes.

The AHRS has two independent channels. Evaluated power of each channel is 3.5 %, of reactor rated power, while maximum power is 7 %. Each AHRS channel ensures heat removal from the reactor without violating the fuel element safety limits set up for design basis accidents.

Each AHRS channel (see Fig. 2) consists of heat removal condenser (HRC) cooled by the distillate of RI equipment cooling circuit, which is connected by steam and condensate pipelines with the separator, that is a part of the secondary circuit. On the condensate pipeline, there is a direct operating valve that opens if the steam pressure in the separator increases over the nominal value.

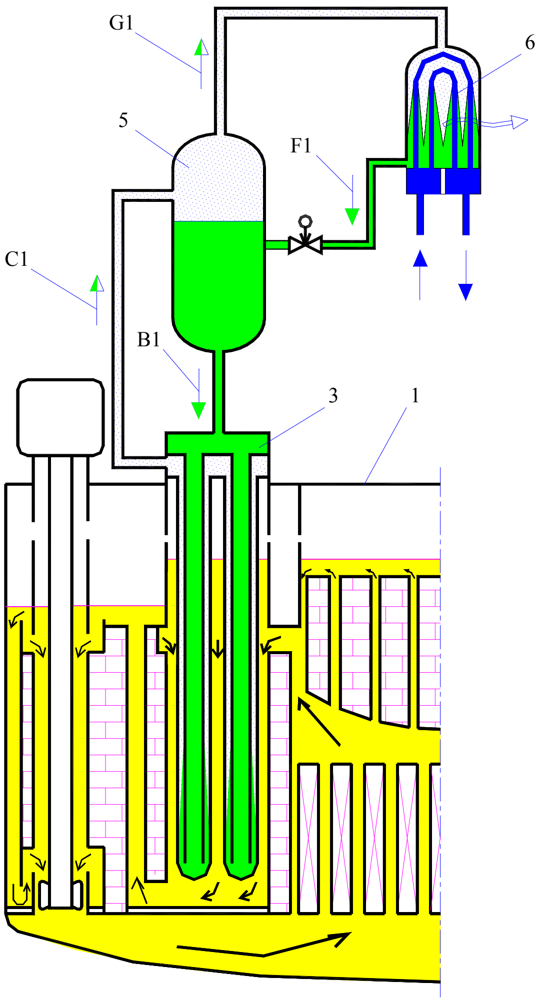


Fig. 2. Autonomous heat removal system

| | |
|---|---|
| 1 – monoblock, | G1 – saturated steam, |
| 3 – SG module, | C1 - -steam-water mixture, |
| 5 – separator, | B1 – boiler water, |
| 6 – heat removal condenser | F1 – condensate drain |

If the RI operates within the normal operational limits, the AHRS is in the stand-by mode, the direct operating valve is closed, and the HRC tube bundle is flooded with a condensate. Heat losses through the AHRS do not exceed 50 kW.

If the separator pressure increases over the specified value, the HRC direct operating valve opens, the condensate begins to drain into the separator and causes HRC tube bundle to be vacated. The secondary circuit steam from the separator begins to condensate on the HRC tube bundle surface, condensate draining back to the separator. This regime of AHRS operation has no time limits.

### 1.5 Passive heat removal system

The PHRS is designed for passive heat removal from the core through the monoblock vessel. The PHRS is a technical means designed for overcoming beyond design accidents with caused by coincidence in any combinations of such postulated events as total de-energizing of the power unit, MCP shutdown, failure of all operating heat removal systems including the AHRC, failure of the reactor safety systems (SS) and failure of the monoblock basic vessel.

The PHRS (see Fig. 3) includes the water tank where the monoblock is installed, the multiple-sectioned heat exchangers installed in the tank and the circuit of cooling water flow connecting these heat exchangers. In case of normal operation of the reactor, the PHRS being in the stand-by mode removes the heat (which is transferred from the monoblock vessel) to the heat exchangers. Power loss is ~0.2 % of the nominal value.

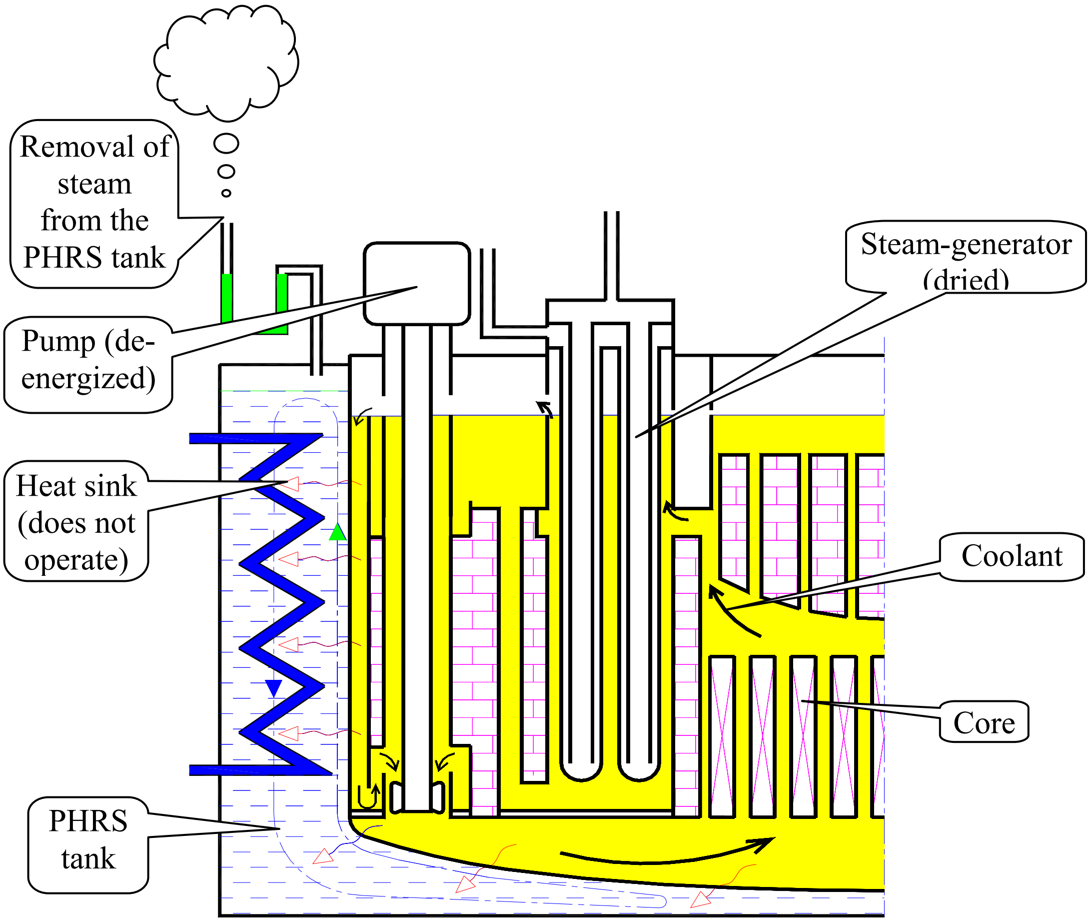


Fig. 3. The PHRS operating scheme

In an event of an accident that is beyond the design basis and is caused by failures in normal operation and safety systems, the lead-bismuth coolant's (LBC) temperature begins to increase. In the course of heating, the monoblock vessel heat losses increase that causes rise of the water temperature in the PHRS tank. Hence, the water temperature in the tank of the PHRS increases up to 100 °C and the power is removed to the atmosphere by the water evaporation. The grace period is 4 days. The damage of the fuel elements does not happen.

Due to the PHRS and monoblock guard vessel, coolant losses that may occur in an event of the beyond design basis accident caused by failure of the monoblock basic vessel are eliminated, thus assuring the reliable heat removal from the core. In this case, the leak is self-confined within the guard vessel.

The volume of the space between the monoblock basic and guard vessels determines the maximum possible coolant loss that may be caused by failure of the basic vessel. In this case the coolant level does not lower below the limit value that assures the conditions of LBC natural circulation in the reactor.

Therefore, in an event of the beyond design basis accident, the PHRS is functioning as both protection and confinement safety system.

### 1.6 Heating system

The system of external steam heating is used for the following:

- initial heating the monoblock up to 180-200 °C before filling it with liquid LBC;
- maintaining the required level of temperatures not less than 160°C in the circuit under conditions, when heat decay energy is insufficient or lacking (during refueling and in the stand-by and technological regimes);
- unfreezing the circuit in case some of the circuit's sections with LBC or the whole volume of LBC are solidified.

The reliability and efficiency of this heating system is verified by operating experience of RIs with LBC at the nuclear submarines (NS).

The heating system includes:

- sections of the steam heating coiled pipes installed on the external surface of the monoblock's basic vessel;
- outside collectors for supply and removal of steam, valves connected by pipelines with the heating steam source, i.e., the start-reserve boiler room and technological condenser.

## 2. Selection of the primary and secondary circuits' coolant parameters.

A conservative approach was used to design RI SVBR-75/100. This approach presumed that the technical solutions borrowed or scaled with small coefficients from the NS RIs were used in the reactor design. Also, these technical solutions have been verified by operating experience of NS RIs and other RIs.

The conservative approach is also characterized by use of the mastered mode parameters of the primary and secondary circuits and orientation to the existing fuel infrastructure and technological opportunities of machine-building enterprises.

Such approach makes it possible to reduce considerably the technical and financial risks, narrowing down the possibility of having errors and failures, which are typical at implementation of innovative nuclear technologies, while significantly reducing the bulk, the execution schedule and the cost of the R&D.

For lead-bismuth cooled reactors, the reliability of fuel elements is in many respects defined by maximum cladding temperature. In accordance with previous R&D, corrosion resistant steel has been selected for the SVBR-75/100 design, corresponding to the maximum temperature of 600°C. Short-time increases of the fuel element cladding temperature up to 800°C without damage are permitted.

The SG operates according to a multiple natural circulation scheme to produce saturated steam; this offers the best lifetime and operating characteristics, e.g., a reliable reactor operation at any power level, simplicity in maintenance of the liquid state of lead-bismuth coolant at low power levels, and simplification of the water chemistry regime of the SG.

Major operating parameters of the primary and secondary circuits' SVBR-75/100 reactor installation are given in Table 1.

Table 1. Major operating parameters of the primary and secondary circuits [1]

| PARAMETER | VALUE |
|---|---|
| Rated thermal power, MW | 280 |
| Maximal temperature of the fuel element’s cladding, °C | 600 |
| Lead-bismuth coolant temperature, °C: at the core outlet/inlet | 482/320 |
| Mode of the secondary circuit coolant circulation (in the separator – SG section) | Natural circulation |
| Steam production rate, t/hour | 580 |
| Steam parameters: pressure, MPa /temperature, °C | 9.5/307 |
| Feedwater temperature, °C | 241 |
| Repetition factor of the secondary circuit coolant circulation | 3 |

## 3. Fuel cycle concept development.

The design of the SVBR-75/100 RI allows it to operate using different types of fuel and in various nuclear fuel cycles (NFC), without changing the RI design or deteriorating the safety characteristics [2].

Once-at-a-time infrequent whole core refuelling that is adopted in the design makes it possible to change considerably the fuel load characteristics in each subsequent refuelling and to use the type of fuel that is most economically effective at a given stage of nuclear power development.

Five variants of the core were considered, different in fuel types; they were as follows:

1. Uranium dioxide, $UO_2$, with an effective density of $\gamma_{eff}$ = 9.65 g/cm$^3$; hereinafter, “effective density” refers to fuel composition homogenized over internal volume of the fuel element cladding;
2. Vibro-packed MOX fuel, $PuO_2$+$UO_2$, with the addition of depleted metal uranium (10 % by weight); $\gamma_{eff}$ = 9.7 g/cm$^3$;
3. Another variant of MOX fuel, including minor actinides such as Np and Am; this composition is referred to as TRUOX fuel;
4. Uranium mono-nitride, UN, with the density $\gamma_{eff}$ = 12.5 g/cm$^3$;
5. A mixture of plutonium and depleted uranium mono-nitrides (PuN+UN); $\gamma_{eff}$ = 10.9 g/cm$^3$. Such fuel composition with a low effective density was selected as a result of reactivity vs. burn-up calculations because it assured the smallest reactivity change during the lifetime.

The isotopic content of plutonium used in the calculations of the abovementioned variants 2 and 5 approximately corresponded to that in light water reactor (LWR) spent fuel. The total quantity of plutonium and minor actinides in variant 3 was taken in accordance with the data of [2], corresponding to LWR spent fuel after long cooling (~15 years). The data on isotopic content of the plutonium fuel compositions are summarized in Table 2.

The power profile along core radius was shaped to flatten power distribution in all considered variants. The radial non-uniformity of power distribution is reduced by changing the content of fissile material in the fuel, which increases from the core centre to the periphery. The maximal radial power peaking factor $K_r^{max}$ was less than or equal to 1.25 in all calculations described below.

Table 2. Isotopic content of Pu and minor actinides in fuel compositions (atomic %)

| Isotope | $^{238}Pu$ | $^{239}Pu$ | $^{240}Pu$ | $^{241}Pu$ | $^{242}Pu$ | $^{237}Np$ | $^{241}Am$ | $^{243}Am$ |
|---|---|---|---|---|---|---|---|---|
| Variants 2, 5 | 1 | 59 | 22 | 13 | 5 | - | - | - |
| Variant 3 | 1.6 | 51.5 | 21.4 | 6.6 | 5 | 5.5 | 7.4 | 1 |

The lifetime calculations were performed for the five variants highlighted above. For variants with uranium fuel (1, 4), the lifetime duration was presumed to be 2200 effective full power days (EFPD); for variants with plutonium fuel (2, 3, 5), the lifetime duration was presumed to be 3200 EFPD. The $K_{eff}$ changes over the lifetime are shown in Fig. 4.

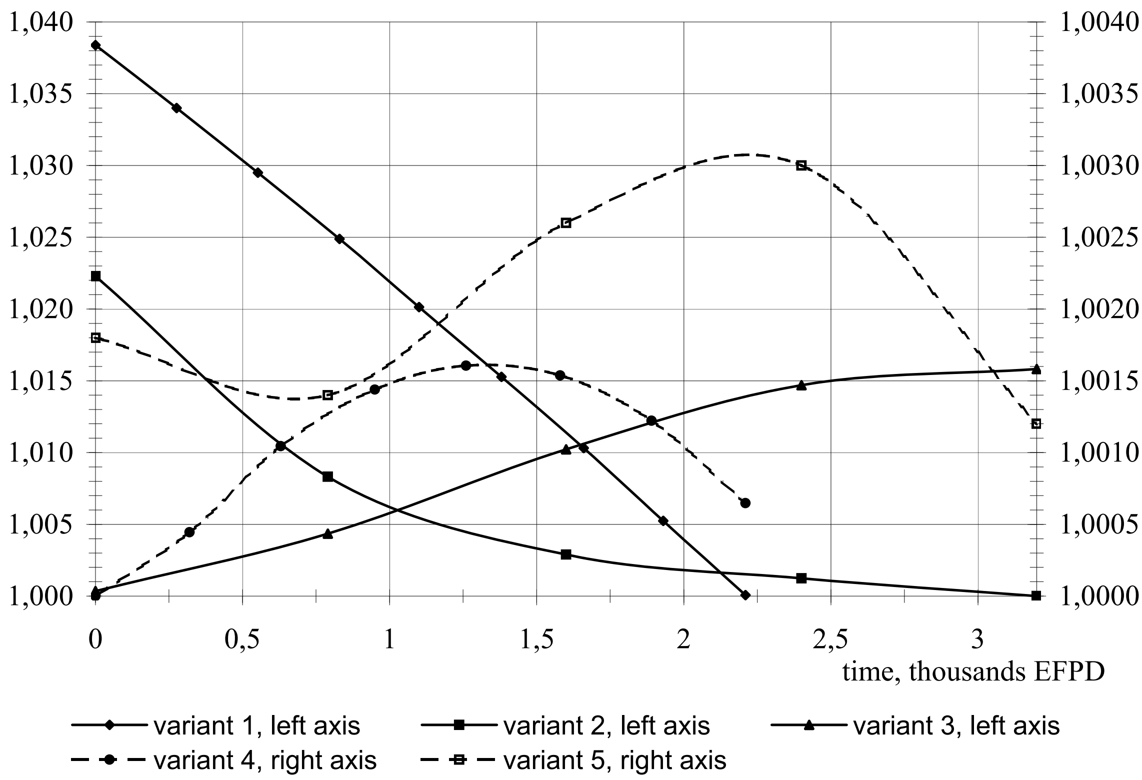


Fig. 4. $K_{eff}$ as a function of time for different types of fuel load.

In the nearest future the use of mastered uranium oxide fuel and operating in the opened fuel cycle with postponed reprocessing will be most economically effective.

Changeover to the mixed uranium-plutonium fuel and to the closed NFC, with CBR≥1 (CBR – core breeding ratio) will be economically effective when the cost of natural uranium increases. At this, the expenditures for constructing the factories for reprocessing the spent nuclear fuel (SNF) and re-fabrication of fresh fuel with plutonium, and their operating costs must be less than the corresponding costs of natural uranium, its enrichment, the cost of manufacturing the fresh uranium fuel and the cost of long-term SNF storage.

At this, it is possible to use both MOX fuel with weapon and reactor plutonium and mixed nitride fuel in case the use of the latter is more economically rational.

Fast reactors (FR) operating in the opened NFC using uranium fuel consume much more natural uranium as compared with thermal reactors (TR). For the planned pace of high nuclear power (NP) development the resources of cheap natural uranium may be expired prior to the middle of the current century. These will cause the increase of the uranium cost and therefore the period of FR operating in the opened nuclear fuel cycle should be maximally reduced.

However, it is difficult to predict reliably the date when the NP will not be competitive with electric power using fossil fuel due to increase of its cost. This is conditioned by a fact that the cost of electric power generated by a nuclear power plant (NPP) is less sensitive to the natural uranium cost in contrast to the cost of electric power generated by thermal electric power plants using fossil fuel.

At the same time the available resources of natural uranium increase progressively at increasing its cost and the rate of carrying out geological and exploration works. This is proved by a fact that for two years the economically available resources of natural uranium in Russia increased from 500 thousand tons to over 1 million tons.

Changeover to the closed NFC will be cheaper in case plutonium extracted from the own FRs SNF of uranium loads is used to form the starting MOX fuel loads of FRs.

When operating by using oxide fuel, a comparatively high breeding ratio (BR) of reactor SVBR-75/100 (~0,84) set conditions for sufficiently large plutonium content in the SNF, which can be used in the next fuel lifetimes while organizing the closed NFC.

Moreover, in the own SNF of starting oxide uranium fuel loads there is much of unburned uranium-235 that is expedient to use for forming the next lifetime load.

That approach to organization of the fuel cycles with whole reprocessing of the own SNF will reduce considerably integral consumption of natural uranium and as for this, it will make the NPPs based on RIs of the SVBR type quite competitive with the NPPs based on RIs with TRs.

As the computations have revealed, changeover to the closed NFC of FR SVBR-75/100 is possible to be begun after the second lifetime i.e. in 16 years. At this, during the first 16 years the total consumption of natural uranium calculated for 1 GWe will be ~5670 tons. During the 60 years of the RI service lifetime, the consumption of natural uranium calculated for 1 GWe will be by 40 % lower than its consumption by WWER-1000 during the same period.

Further FR operating in the closed NFC prior to reaching the equilibrium mode of refuellings will be realized practically without consumption of natural uranium. Comparison of natural uranium consumption by 10 reactors SVBR-75/100 at proposed changeover to the closed NFC and that by one reactor WWER-1000 operating in the opened NFC is shown in Fig. 5.

Use of plutonium extracted from the TR SNF to form the starting loads of the SVBR-75/100 reactors for a purpose of entire elimination of natural uranium consumption in the very beginning of realizing the wide program of implementation of such reactors in the NP will be more expensive as compared with a considered variant of changeover from the opened NFC to closed NFC.

This is conditioned by the fact that for plutonium extracted from the TR SNF the plutonium cost determined by a scope of SNF reprocessing calculated for 1 t of plutonium will be several times higher as compared with its cost in case the own SNF is used due to much less content of plutonium in the TR SNF.

It should be taken into account that organization of large-scale reprocessing of TR SNF and manufacture of MOX fuel should precede the construction of NPPs with FRs. Thus, the demands in investments increase.

At the same time for the proposed changeover from the opened NFC to the closed NFC the organization of the closed NFC may be long postponed from the moment of FR launching that diminishes the investment demands. At this, as computations have revealed, the investment fund necessary to organize that may be formed for approximately two years due to including the corresponding component into an electricity cost after ending the pay-back period of the NPP at keeping the economic efficiency at the same level.

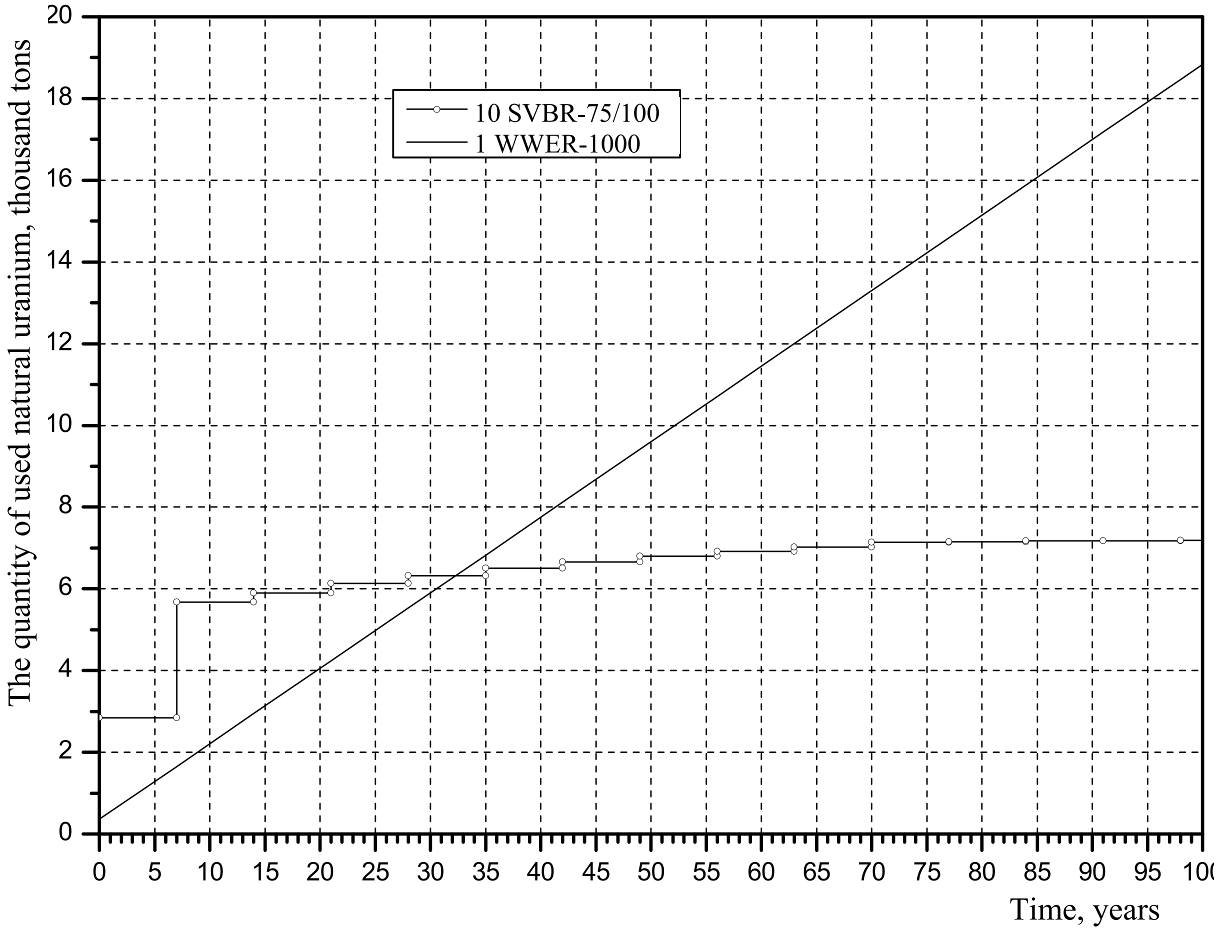


Fig. 5. Integral consumption of natural uranium at power of 1 GWe

With due account of the own investment potential of the NP, for the proposed variant of NFC organization at the same extent of annual investments it will be possible to introduce more FR capacities.

In the closed NFC when MOX fuel is manufactured, instead of waste pile uranium, the TRs' SNF (of both WWER and RBMK) may be used (utilized) without separating uranium, minor actinides and fission products (FPs) similarly to the DUPIC technology developed for reactors CANDU. In this case after the gas and volatile FP have been eliminated, the TRs' SNF is replaced by depleted uranium in MOX fuel. The scheme of that fuel cycle is shown in Fig. 6.

Adaptability of reactor SVBR-75/100 relative to the fuel type and fuel cycle makes it possible to realize timely and gradual changeover to the closed NFC, which will be economically justified. Simultaneously, that solves a problem of utilizing thermal reactors' SNF and radiation-equivalent burial of long-lived radioactive waste (RAW), taking into account that minor actinides are effectively burned in the FR.

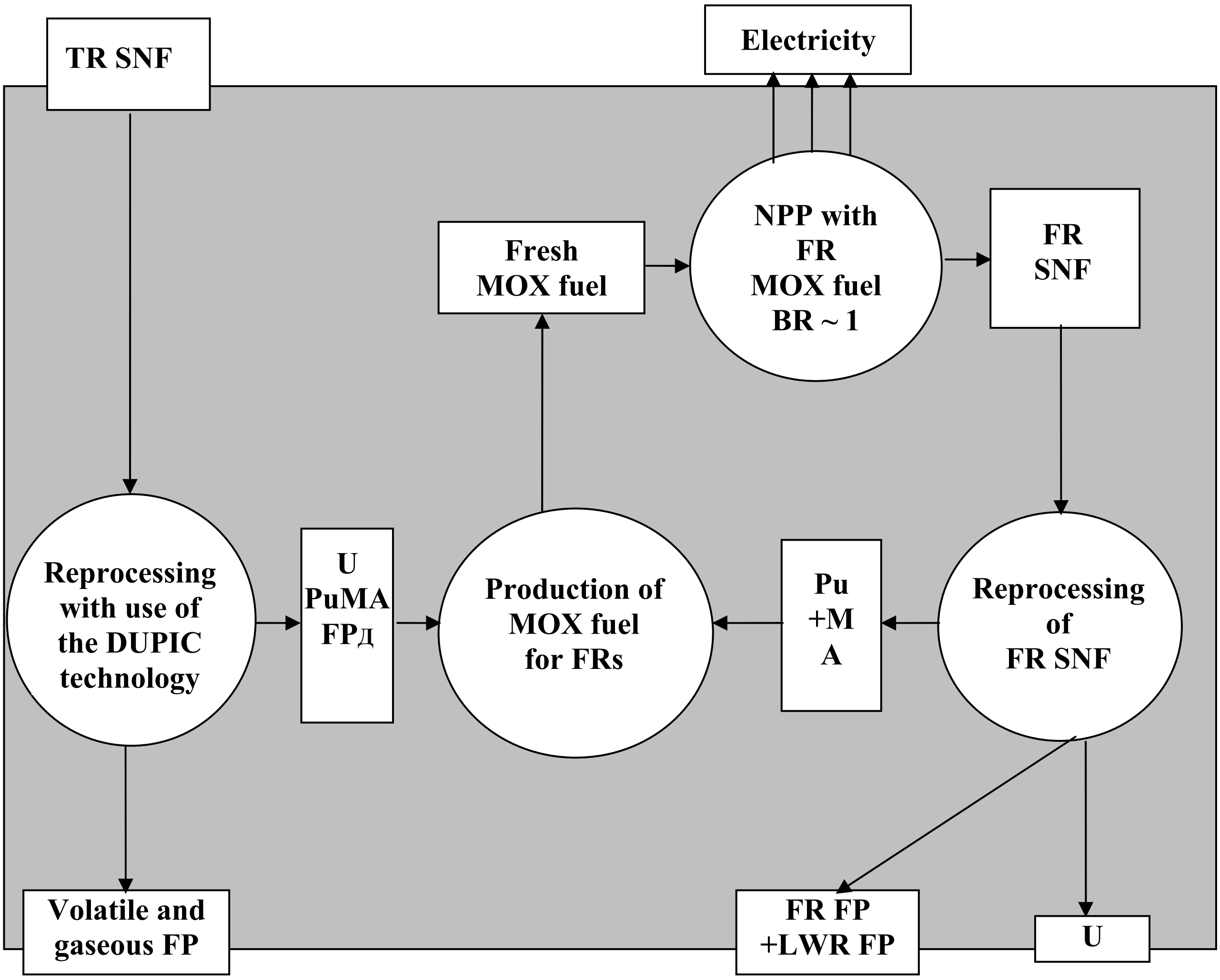


Fig. 6. The principal scheme of direct utilization of LWR SNFThe flexibility of the SVBR-75/100 in relation to fuel cycle technologies is realized in compliance with the principle: "To operate using the type of fuel and fuel cycle that are most efficient today" makes it possible to postpone the construction of specialized fuel cycle factories for several decades after the first NPP unit with the SVBR-75/100 modules is launched. For example, after the introduction of about 10 GW(e) using the SVBR-75/100 and repaying the NPP construction costs, a share of the profits could be spent to develop the industry for spent fuel reprocessing and MOX fuel fabrication.

For reprocessing of the SVBR-75/100 spent nuclear fuel, it is presumed that the extracted fission products are vitrified and, after necessary cooling and enclosure in special containers providing a multi-barrier shielding, they are transported for final disposal in deep geological formations. Minor actinides (except curium) are not separated from plutonium and are used in the reactor as a fuel component. Due to the high power release caused by alpha decay, curium is extracted and transported to the temporary repository for 100-year cooling. After cooling, all curium isotopes (except curium-245) are transformed into plutonium isotopes, and this isotopic mixture is transported back to the reactor for further burning.

## 4. Nonproliferation concept development.

Nonproliferation of fissile materials means creating the conditions when inappropriate use of fissile materials is least attractive for potential proliferators of the nuclear weapon.

The solution to the problem of non-proliferation can be only achieved by coupling both technological and political measures. The relationship of those measures will be different for nuclear and non-nuclear countries. During the recent decades all nuclear countries, which are the members of the "Nuclear Club" and legally possess the nuclear weapons, have solved this problem successfully, using the measures of physical protection, accounting, control and safeguarding. For that reason, the additional measures of technological maintenance of non-proliferation will be justified if they do not reduce the NP competitiveness.

When using NPPs in developing countries, the additional measures of technological maintenance for non-proliferation should be implemented, together with political measures and international control [3].

Compactness of the module and LBC properties provide a unique opportunity to realize return of the SNF without its unloading from the reactor in the User-Country. Transportation of the fuel in the reactor monoblock with solidified LBC creates an additional technical barrier for the theft of fuel. The solidified LBC in the monoblock eliminates a risk of nuclear and radiation accident while transportation is performed.

In all these applications, the technological support for nonproliferation is also assured by the following features. When uranium fuel is fabricated, uranium enriched in less than 20 % will be used. At the stage of SNF reprocessing, 2 % of fission products built-up in the SNF and all minor actinides (MA) will remain in the re-fabricated fuel, except for curium that is released and kept to decay in plutonium with return to the fuel cycle. Handling that fuel needs the especial technological equipment, which makes it easy to account and control the fuel movements. Breeding zones in which plutonium for weapons can be built-up are also absent in the reactor.

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