BN-800 – HISTORY AND PERSPECTIVE

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ABSTRACT

The sodium cooled fast reactors are one of the most developed and advanced directions of future nuclear engineering. Russia is the first among other countries in field of fast reactor development.

The idea of fast reactor designing was proposed in the former Soviet Union by Dr. A.I. Leipunski at the end of 40th. The successful operation of Russian fast reactors (BOR-60, BN-350 and BN-600 [1]) and the world experience proved the feasibility, reliability and safety of this direction of nuclear engineering and allowed to begin the development of the BN-800 reactor project as the commercial fast reactor.

In 1992 Russian Government confirmed the construction of BN-800 reactors on South Ural NPP in Chelyabinsk region and on Beloyarskaya NPP.

This report presents the brief review on main directions of BN-800 reactor development carrying out in IPPE

HISTORY OF BN-800 DESIGN

The first design [2] of BN-800 reactor was developed and was undergone an examination in 1985. It fulfilled the demands made of the reactors in that time.

But last time (after the serious accidents on Chernobyl NPP and Three Mile Island NPP) the aspects of safety increase of NPP play a leading role when designing the new reactors serving the economic competitiveness.

All these aspects were introduced into the new Nuclear Safety Rules [3], adopted in our country in 1989. This Rules include the requirement of guaranteeing of negative reactivity coefficients on reactor power and coolant temperature.

After having adopted the new Safety Rules the commission of Russian Academy of Science headed by Dr. V. Subbotin made new examination of BN-800 reactor project. The commission noted the large positive value of sodium void reactivity effect (SVRE) as a main disadvantage of this project. The recommendation was to develop a new reactor core design with negative value of SVRE.

The first investigations in the end of 80th showed the principal possibility to achieve zero (or negative) SVRE value in the reactor core by introduction of sodium plenum above the core [4].

The analysis of the numerous ways to reduce the SVRE value allowed to choose the most optimal core design [5].

The design project [6] of BN-800 core was developed in 1992. During next 5 years the complex justification of reactor physics was carried out based on calculational analysis and experimental investigations at the BFS facility critical assemblies [7] and based on results of international benchmark analysis of BN-800 reactor core with sodium plenum [8].

At the end of 1998 we got the license for reactor construction.

PHYSICAL RESEARCH AT CRITICAL FACILITIES

During 1993-1997 at the critical facility the experiments on full-scale simulation of the BN-800 reactor with a sodium plenum were carried out [9,10,11]. As a mock-up the critical assembly BFS-58 was used which most fully (under the BFS-58 facility conditions) represented main features of the new BN-800 reactor core design. The main object of experiments was to study the sodium void effect of reactivity at removal of sodium from the sodium plenum, the core, control rods sleeves. Besides the sodium void reactivity effect there were measured fission reaction rate distributions for some isotopes over critical assemblies height and radius and control rod efficiencies as well.

The worst condition of the reactor from the viewpoint of the sodium void effect that is reached by the end of the cycle under stationary refueling conditions was simulated.

In distinction to previous assemblies in this one the low enrichment zone was fully made of uranium-plutonium fuel that ensured rather a representative volume in which the value of the sodium void reactivity effect is positive.

The results of experiments at the BFS-58 critical assembly are as follows:

1) The experiments of the sodium void reactivity effect have shown the same trend of discrepancy between the calculation and experiment as at the BFS-54 and BFS-56 critical assemblies; the diffusion calculations overestimate the sodium void efficiency predicting a more negative SVR value in it than in experiments; in the core, on the other hand, the calculations predict a more positive SVR value than in

experiments. Quantitative estimates of these discrepancies are as follows:

- in the sodium plenum the calculated value obtained by the Monte-Carlo method practically is in agreement with the experimental one; the diffusion calculations differ from experiments for a case of maximum void area by a value of 0.1-0.2 %Δk/k;
- in the core – the calculated SVR value for a case of maximum void area exceeds the corresponding experimental value by a value ~0.25%Δk/k for ABBN-78 nuclear constant library and ~0.1Δk/k for ABBN-93 nuclear constant library. One should note good agreement between calculated values obtained by various codes (diffusion theory and Monte Carlo method).

2) In addition to experiments on sodium removal from the core and sodium plenum the experiments on sodium removal from control rod sleeves were carried out. These experiments have shown that in the critical assembly under consideration this reactivity effect has a negative sign and is well predicted by calculation by Monte Carlo method. Calculations by diffusion codes, as it was expected, overestimate too much (by the absolute value) the value of the effect.
3) When extending the results obtained to the reactor two factors should be taken into account; differing critical assemblies from reactor. The first factor is the modelling on the basis of weapon-grade plutonium but not civil for reactor. The second difference consists in that in modelling fission products were absent, which were modeled by uranium-238. The both factors can be taken into account in traditional way, that is by uncertainty estimate of considered functionals (SVRE) calculation reasoning from uncertainties of neutron cross-sections for Pu-240, Pu-241, Pu-242 and fission fragments.

The studies performed have shown that calculation uncertainty of SVRE in transition from power plutonium to weapon-grade plutonium does not exceed 30% rel. Taking into account that SVRE value in such transition decreases ~0.15%ΔK/K, maximum uncertainty value in the effect will not exceed ±0.05%ΔK/K. Uncertainty of calculation displacement for SVRE value (calculation–experiment) due to lack of fission products in the model does not exceed 20% rel. or ±0.1%ΔK/K.

Calculations of void effect for BN-800 reactor in a state “before reloading” (cycle end) using the codes and constants mentioned gave for SVRE value:

$$-0.21\pm0.16\%\Delta K/K\ (2\sigma)$$

The results of experiments carried out at critical assemblies and their extension to reactor accounting for the above factors allows to consider that actual SVRE in BN-800 reactor core will be equal: **–0.46±0.28 %ΔK/K**(2σ) [12]. Maximum uncertainty value was obtained by square summing up of the above mentioned uncertainties. Even in case of maximum calculation error realization SVRE value will be equal to ~-0.18%ΔK/K.

UTILIZATION OF EWAPON-GRADE PLUTONIUM

Calculations performed have shown that the utilization of weapon-grade plutonium in the BN-800 reactor though leading to some changes in physical characteristics of the reactor but does not appreciably affect the adopted design operating conditions of the reactor operation on the reactor-grade plutonium. A significant decrease of initial fuel enrichments when passing over to the weapon-grade plutonium at fuel fabrication is realized by using the provided and developed method of initial enrichment correction. Some decrease of the breeding ratio (~0.07) in the core results in an increase of the reactivity loss during the inter-refueling operation period (~13% relative) and, consequently, in an increase of reactivity margin. However, these changes do not violate the requirements of NSR RU AS-89 on safe operation of the reactor. It is ensured due to some margin available in the design-version control and safety rods efficiency, as well as due to some increase (~4% relative) of their efficiency at passing over to the weapon-grade plutonium. One of the most important (from the viewpoint of safety) characteristics, the sodium void effect of reactivity, becomes more negative at passing over from the reactor-grade plutonium to the weapon-grade one. Calculations have shown that the replacement of the reactor-grade plutonium with basic isotopic composition (**Pu239/Pu240/Pu241/Pu242=60/25/10.9/4.1%%** respectively) by the weapon-grade one reduces the sodium void effect reactivity by **~0.2%Δk/k** [13].

BN-800 REACTOR DESIGNS FOR NUCLEAR WASTES UTILIZATION

The most pressing problem associated with the use of nuclear power is the accumulation of high level waste in the form of fission product and actinides. The letter are dominated by plutonium and a group of so-called minor actinides, like neptunium, americium and curium. Possibilities for addressing the HLW problem that are currently under consideration include 1) deep geological storage and 2) incineration of the actinides and long-lived fission products in reactors.

The first solution, despite its apparent is not easy. A quite well-founded distrust exists when it comes to subjects such as, waste form integrity during storage, stability of rock formations, possibility to provide sealing of inlets, possibility to prevent water intrusion from adjacent rocks, etc., due to the time scales involved. It therefore seems that second solution may provide a more reliable control method than opposed to the first.

Fundamental features of fast reactors, linked to the neutron cross sections of the actinides for high energy

neutrons, allow for effective burning. Irrespective of breeding fuel the introduction of fast reactors into nuclear power system will permit the carrying our of effective burning of the actinides. In the overall system, the fast reactors function primarily as waste burners while the thermal reactors function primarily as electricity producers. Some of the design characteristics of conventional fast breeder reactors needed to be converted to better conform to the actinide burner reactor role. The conversion requirements have been identified at a conceptual level and include the following [14]:

– replacement of breeding blankets for non-breeding;

– increasing of fuel enrichment;

– using of fuel without uranium-238, replaced with inert matrix.

Replacement of breeding blankets by non-breeding does not require a solution of any new problems and is determined only by technical possibilities of a particular reactor. Well known materials, used in the fast reactor technology – stainless steel and boron carbide with natural enrichment – can be considered as candidate materials for non-breeding blankets. Replacement of axial breeding zones, included in fuel element, has more technical restrictions in comparison with the replacement of radial breeding zone, but, in principle, here can be found acceptable technical solutions both for the operating BN-600 reactor and for BN-800 reactor design [15].

A more complicated problem appears in the case of increasing the fuel enrichment. Oxide fuel, which is considered first for actinides burning as well mastered and investigated, has a limit in enrichment, determined by its solubility when chemically treated. The data available suggest a hope to reach enrichment level of 40 – 45 %, though this requires a technological substantiation. This enrichment level provides already high burning characteristics. As an example, fast reactor with such fuel enrichment 1 GW in power provides burning of 500 kg plutonium per year (at φ=0.8). What is the manner in which the fuel enrichment can be increased? The following principle ways exist [16]:

- introduction of absorbing fittings into the core;
- a decrease in fuel volume fraction;
- introduction of absorbing blankets.

An engineering optimization permit to find the most appropriate variant both for the operating fast reactors and for reactors under design.

The highest actinide burning efficiency can be obtained using the fuel without uranium-238. Reactor with such fuel (N_{el} =1GW and φ=0.8) can provide a limit plutonium burning level of ~ 750 kg/year. Some fuel compositions without uranium-238 are under developed on the base of inert matrix – magnesium oxide, circonium carbide, aluminium nitride etc. Significant time will be of course taken for development and testing of such fuels, before they could be used in fast reactors.

The necessity to use fast reactors in nuclear power is determined not as many by their ability to burn plutonium, but their possibilities to burn the minor actinides. In fast reactors, the minor actinides are subject to fission by high-energy neutrons, that is they can be used as a nuclear fuel. The simplest way to burn the minor actinides is to mix them with the bulk fuel. The addition of minor actinides to the fuel leads to a noticeable increase in SVRE value. That is why this process can be organized using only cores with an increased fuel enrichment which have some margin in a negative SVRE value.

Another way to burn the minor actinides is considered – in special subassemblies (SAs) with a high concentration of the minor actinides located into an inert matrix.

Here, it is appropriate, for different reasons, to use for burning a separated minor actinides mixture, that is to burn separately americium and neptunium, and curium, which forms only ~6 % of the total mixture, should not be burned at all, because it, possessing high neutron capture, transits into successive high-level isotopes.

This requires of course the creation of a special production, including a complete cycle– from fuel pin production to their reprocessing, but of smaller scale as compared with the existing production. The calculation studies show, that a heterogeneous introduction of the minor actinides into the core does not solve the SVRE problem; from this standpoint, the best option is location of specialised subassemblies in the radial blanket.

Even more intensive burning of the minor actinides is possible in a specialised fast reactor (or specialised core), the fuel of which contains a considerable amount of minor actinides.

In a specialised reactor it is appropriate to use fuel on the base of inert matrix without uranium-238 which is a supplier of the plutonium group and minor actinides. Optimization studies show that such fuel can include up to 30÷40% MA in a mixture, for example, with uranium-235 (limiting parameters here are SVRE, β_{eff} and some others). Several such reactors, each being capable to burn 500 kg MA per year, can service the entire nuclear power in the country.

A) Transmutation of HLW in special devices

Burning of radioactive wastes of the nuclear power (minor actinides and fission products) in special irradiation devices located outside the core has some essential advantages [17]: the effect of these devices on the core neutronic parameters decreases greatly, a possibility for long-term irradiation appears etc. However, the irradiation process in these devices will be meaningful only if a radiotoxicity of the remaining after irradiation nuclides is essentially less than that of initially introduced wastes. All this imposes the restrictions on burning process.

Let us consider the aspects connected with transmuting process of minor actinides. In this case the irradiation device may contain americium, since its quantity exceeds essentially the accumulation of other dangerous nuclides.

Fig.1 presents the radioactivity change of irradiation devices with americium after irradiation for different americium burn-up.

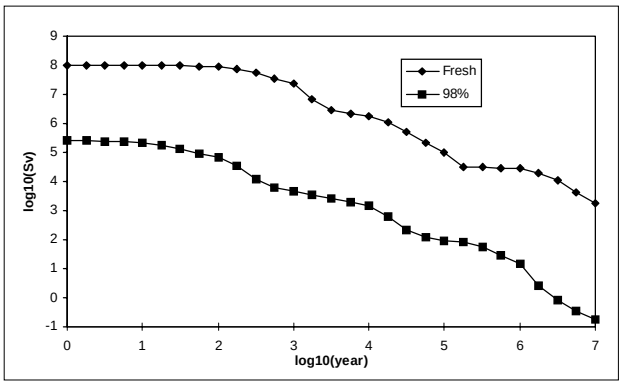

FIGURE 1. Radiotoxicity change for non-irradiated and irradiated americium

One can see that only a very deep target burn up (more 90%h.a.) with allow a decrease in waste radiotoxicity, as compared with the case when all americium is wasted, and for essential waste radiotoxicity decrease, a burn-up more than 95%h.a. should be reached.

What is the way to reach so high burn-up in irradiation devices located in the radial blanket, where a neutron flux is much less, as compared with the core? The simplest way – an increase in irradiation time. Unfortunately, to reach this burn-up, the irradiation time should be more than 50 years. Clearly, in the conditions of modern reactor technologies this situation is not realistic.

However, if to pay attention to the fact that the actinide interaction cross-sections in a thermal spectrum are one order higher, as compared with fast spectrum, an idea appears to develop irradiation devices with a large moderator content.

The detailed calculation studies have shown a principal possibility to reach high americium burn-up with irradiation time ~10 – 15 years. True, it requires that the moderator fraction in the irradiation devices be ~10 times higher the loaded americium fraction. In this case, for example such irradiation devices (90 items), loaded to BN-800 reactor radial blanket, would allow to burn-up to 60kg americium, per year thus solving the problem of actinide radiotoxicity decrease. Notice that this americium quantity is accumulated annually in approximately seven VVER-1000 reactors, and for utilization of americium accumulated in the nuclear power of Russia (installed power 21.2 GW(e)), three BN-800 reactors with a modified radial blanket would be needed.

Similar irradiation devices can be used also for efficient utilization of the most long-lived fission products, such as Tc^{99}, I^{129}, Pd^{107},Cs^{135} etc. These waste quantity in spent fuel of nuclear reactors in also essential. For example, approximately 26 kg Cs^{135}, ~18 kg Tc^{99}, ~11 kg Pd^{107} etc will be unloaded annually from BN-800 reactor.

Surely, the radiotoxicity of fission products is much lower than the radiotoxicity of minor actinides, however, from the other side their chemical properties allow them to migrate in the earth more quickly than actinides. This aspect is a motive for performance the studies in fission product transmuting possibility.

A homogeneous recycling of fission products in BN-800 reactor fuel allows an annual destruction not more than 5% of loaded quantity, which from economical standpoint is hardly advisable. A homogeneous recycling in thermal reactor fuel is impossible because of essential degradation of the reactor neutronic parameters. Therefore, the most acceptable method for fission product transmuting remains their burning in irradiation devices located in the radial blanket. Taking into account the fact that the capture cross-sections for practically all fission products have a maximum in the thermal or close to thermal region, the moderator introduction to irradiation devices, similar to americium case, would allow an essential increase in the fission product transmuting efficiency. The studies performed have shown that when introducing the moderator quantity 10 times more than that of loaded fission products, one may reach 80% transmuting of Tc-99 and Pd-107, 70% transmuting of I-129 and 50% – Cs-135. The activity change of irradiation devices regarding non-irradiated targets as a function of time is presented in fig.2.

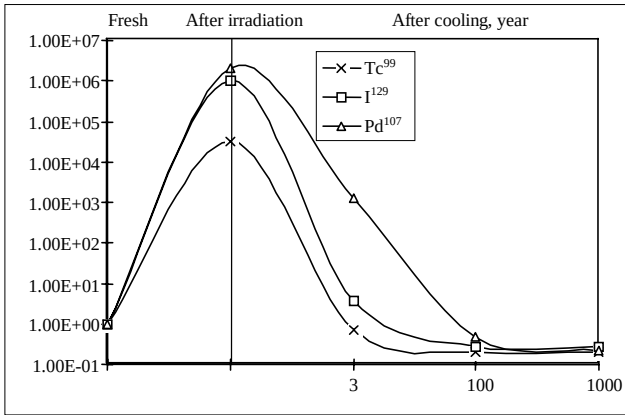

FIGURE 2. Irradiation devices activity change regarding non-irradiated targets

A splash of the irradiation device activity during irradiation is defined by creation of long-lived nuclides. The activity of these nuclides decreases rather quickly, and in 30-50 years the irradiation device activity will be defined by a non-burned part of the initial fission product. Notice that with this burn-up an absolute quantity of transmuted fission products will be small due to their low loading into irradiation device.

The increase in fission product loading for the sake of decrease in moderator fraction leads to increase in absolute consumption of fission products, however, in this case the transmuted fission product fraction decreases. This fraction decrease leads to that a rather large quantity

of FR remains in the irradiation device, and repeated reprocessing of these devices will be needed for subsequent involvement of the remaining fission products into the next irradiation cycle. This reprocessing can lead to an essential increase in non-returned losses.

Thus the choosing one or other scheme for fission product transmuting should be dictated by the chosen strategy for nuclear power fuel cycle.

The considered aspects of utilization special devices, containing large quantities of moderator, for the purposes of efficient transmuting the radiation wastes of the nuclear power have shown a principal possibility for decreasing these waste radiotoxicity. However, this requires a large time of irradiation in these devices (10-15 years), and by now the possibilities of existing reactor materials to operate in these extreme conditions have not yet proved.

B) Closed fuel cycle organization with BN-800 reactors with inert matrix fuel

By now rather wide studies have been carried out on analysis choosing different fuel compositions without uranium-238 replaced by an inert matrix. Compositions based on zirconium carbide? aluminium nitride, magnesium oxide etc are considered [18]. In the given studies, a composition based on magnesium oxide was considered. A core based on fuel without uranium-238 has some specific features. First and foremost the elimination of uranium-238 provides a negative SVRE value for the core, and this is a doubtless advantage of this type cores [19]. At the same, Doppler-effect value decreases noticeably, which in traditional cores is defined by uranium-238. Partially the Doppler-effect can be restored by introduction into the fuel composition of resonance absorbers, the best of which are iron, niobium and tungsten. The elimination of uranium-238 leads also to essential increase in reactivity loss rate due to burn-up which in turn leads to decreased time between reloading.

However, the most sensitive disadvantage of this type cores is a large difference in power of fresh and spent SA. which increases essentially the power non– uniformity. In designing a core based on fuel free of uranium-238 for BN-800 reactor, the latter problem was solved in the following way: a core consisting of two subzones with different plutonium content (similar to zones of low and high enrichment in traditional reactors), was divided into ten concentrical subzones (five in each subzone with different plutonium content), and a reloading was performed through external subzones by reloading SA from periphery to core centre forming an equalized power field. In this case a 50% fuel burn-up corresponds to the best power field equalization. The increase in fuel burn-up, which is possible for fuels with inert matrices, will lead to increase in the power field non-uniformity and will require a larger core volume. In the core under consideration, a specific feature of BN-800 reactor subassembly design was conserved, consisting in availability of a sodium plenum above the core to decrease SVRE. The fuel of this core may contain up to 15% MA with retention a zero SVRE value. The major BN-800 reactor parameters with fuel free of uranium-238 are presented in table 1.

TABLE 1. Major BN-800 reactor parameters with fuel free of uranium-238

	Parameter	Value
1	Nominal (rated) power, MW(e)	800
2	SA number in the core including: first subzone second subzone	565 145 420
3	Fuel composition	(Pu, MA)$\hat{I}_2$ MgO
4	Fuel effective density, g/cm^3	4.5
5	Actinide (Pu, MA) content in fresh fuel, g/cm^3 (85%Đu, 15%ÌÀ)	1.6
6	SVRE in the core, %Δê/ê	~ 0
7	Doppler-effect in the core, %Δê/ê	-0.235
8	Reactivity change rate, %Δê/ê/1months	3.5
9	Time between reloading, months	3.0
10	Max linear power, W/cm	500

The parameters presented can be improved in choosing the core for specialized reactor. The core variant presented has been chosen reasoning from existing limitations in BN-800 reactor (CSS rod number, core volume etc.).

The BN-800 reactor with the core considered above can fulfill functions of a reactor-burner of plutonium and minor actinides. We consider here a simple model of nuclear power system functioning, consisting of thermal reactors VVER-1000 and fast reactors of BN-800 type [20,21]. In this model BN-800 reactors consume plutonium together with minor actinides, which are extracted from VVER spent fuel and are mixed with plutonium and minor actinides, which are extracted from BN-800 spent fuel. Table 2 presents actinide content in spent fuel of VVER type reactors.

The dynamics of fuel cycle parameter changing was studies for two decay time periods of fast reactor spent fuel: T=1 yr and T=3 yr.

The calculations have shown that in the course of recycling a smooth increase in minor actinide content in the fuel takes place – from ~7% (initial loading of fuel from VVER) to 10-13% in a quasistationary regime. The latter establishes in approximately 25 cycles. The fuel parameters in a quasistationary regime are presented in table 3.

In the nuclear power model considered, one BN-800 type reactor with fuel free of uranium-238 can utilize plutonium and minor actinides produced in 2.6 VVER-type reactors.

TABLE 2. Quantity of actinides produced in spent fuel of thermal reactors, kg/GW(e)

Isotope	Uranium fuel	Uranium-plutonium fuel (30%)
Bcero	235	385
Pu^{238}	2.8	6.8
Pu^{239}	121.7	160.1
Pu^{240}	53.2	96.7
Pu^{241}	27.5	56.9
Pu^{242}	12.8	31.8
Np^{237}	7.1	11.2
Am^{241}	6.4	8.0
Am^{242}	.006	.03
Am^{243}	2.6	9.1
Cm^{242}	–	.05
Cm^{244}	1.0	4.0
Cm^{245}	–	.2

TABLE 3. Established isotope composition of fast reactor fresh fuel, (kg/t)

Isotope	**$T_{вн}$= 1 year**	**$T_{вн}$= 3 years**
Pu^{238}	46.6	53.5
Pu^{239}	249.0	250.1
Pu^{240}	358.7	358.3
Pu^{241}	80.2	67.9
Pu^{242}	118.6	119.0
Np^{237}	17.1	17.8
Am^{241}	47.2	62.1
Am^{242}	3.0	3.5
Am^{243}	46.6	43.7
Cm^{242}	.6	.06
Cm^{244}	28.3	20.1
Cm^{245}	5.3	4.0

The preceding conceding consideration has shown that in the case of repeated recirculation of plutonium fuel with 7% MA through the core with fuel free of uranium-238, BN-800 reactor utilizes ~45 kg MA per year. This corresponds to MA production per year in 2,6 VVER-1000 reactors. In this case MA quantity in the fuel is monotonically increasing, tending to a quasistationary state at a level of ~14%.

How much the MA content in fresh plutonium fuel being loaded to BN-800 reactor can be increased? The answer to this question depends on study results of many factors. The results were presented [20] on radiation parameters and heat release calculations for fresh fuel, which essentially influence the technological process for fuel manufacture. Here we note the most important neutronic parameter – SVRE value. The calculations performed show that for the core considered an allowable MA content in the fuel from the standpoint of providing zero SVRE value amounts ~15%.

The studies in dynamics of a fuel cycle with 15% MA content in fresh fuel indicate that MA level in a quasistationary state will be ~18%. The organization of this fuel cycle will allow the MA utilization per year ~2.0 times more, as compared to the fuel cycle with 7% MA content, that is in this case ~90 kg MA per year will be utilized.

Thus, the variant considered is probably limiting for the core with plutonium fuel. MA burning efficiency can be increased when using the fuel on the basis of uranium-235. The calculations performed show that a zero SVRE value for this fuel is retained at addition 35% MA. In this case the MA burning efficiency increases additionally 1.9 times, as compared with the previous case and will amount ~170 kg/yr. Consequently, BN-800 reactor core with a fuel on the basis of uranium-235 allows to serve from ~10 VVER-1000 reactors.

It is interesting to note that in a quasistationary state the minor actinide fraction decreases up to ~16%. Table 4 present the calculation results for compositions of unloaded fuel at reactor make-up by fuel (85% Pu, 15% MA) and fuel (65% U-235, 35% MA). One can note that the plutonium composition, which is generated in the second case from uranium-235, contains a very large quantity of plutonium-238.

TABLE 4. Unleaded fuel composition at reactor make-up by fuels on the basis of Pu and U-235 (relat.%)

Isotopes	**Fuel**	
	85%Pu, 15%MA	**65%U-235, 35%MA**
U^{234}	0.18	2.90
U^{235}	0.0029	36.96
U^{236}	0.00017	29.99
Np^{237}	7.23	10.11
Pu^{238}	4.65	8.62
Pu^{239}	41.57	1.67
Pu^{240}	24.09	2.35
Pu^{241}	7.41	0.26
Pu^{242}	5.48	0.81
Am^{241}	6.39	2.88
Am^{242m}	0.19	0.15
Am^{243}	2.13	1.94
Cm^{242}	0.0034	0.0013
Cm^{243}	0.026	0.011
Cm^{244}	0.58	1.00
Cm^{245}	0.049	0.15
Cm^{246}	0.0039	0.065

CONCLUSION

Thus on the basis of BN-800 reactor project it is possible to design the universal fast reactor permitting to

solve rather effectively the different problems of nuclear fuel cycle: from high breeding of secondary nuclear fuel to the effective transmutation of long-lived nuclear wastes depending on state of nuclear market.

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