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ARTICLES

MODELS OF THE DEVELOPMENT OF LARGE-SCALE NUCLEAR POWER IN RUSSIA WITH A TRANSMUTATION NUCLEAR FUEL CYCLE AND ATTAINMENT OF RADIATION EQUIVALENCE OF HIGH-LEVEL WASTES AND NATURAL URANIUM

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The availability of resources and the ecological advantages of operating with no serious accidents are motivating studies of the possibility of the development of large-scale nuclear power production in Russia in the next century. The conditions for such development are greater safety of modern reactors and switching in the future to inherently-safe reactors as well as a substantial decrease in the activity of the finally buried wastes, in the limiting case down to .he activity of the uranium employed. The difficulties and the problems of nuclear power are solvable, but the desirability of improving nuclear power production and developing an infrastructure should be tied to a need for a substantial increase (approximately ten-fold) in capacity and the relative contribution of nuclear power to the production of electricity up must be increased to 30% in order to limit the increase in the consumption of fossil fuel in the next century.

The reserves of depleted uranium (200 thousand tons) and natural uranium (600 thousand tons) are sufficient for long-term (3-4 thousand years) operation of large-scale nuclear power with a total capacity of 300 GW(e), which gives 2 kW per capita with internal use of electricity.

This paper examines the fuel supply for nuclear power and the choice of ways to handle long-lived high-level wastes.

Variants of the Nuclear-Power Development. We shall consider the handling of high-level wastes for three possible variants of the development of nuclear power with self-provision of fissioning materials on the basis of the following reactors:

YaÉ-1 — improved safety, burnup and breeding of fuel in modern designs or greatly advanced designs of the near future: fast BN-800 reactors and thermal reactors (AST and others), for example VVÉR-1000 [1, 2] (Fig. 1a);

YaÉ-2 — improved inherently-safe BREST-300 [3] and thermal VVÉR-1000 (both variants of development with substantial utilization of natural uranium) (Fig. b); and,

YaÉ-3 — fast-neutron reactors without utilization of natural uranium in the initial fuel load; in this variant, BN-800 reactors are developed first, their development stabilizes at 1/3 full capacity of the nuclear power production, and the development of second-variant reactors starts (BREST-300 and VVÉR-1000) up to 2/3 total capacity, after which at constant capacity the BN-800 reactors are decommissioned and the final, second-variant reactors are finally put on line (Fig. 1c).

The last two variants of nuclear power, where the BREST-300 reactors, whose safety is based on the laws of nature and in which serious accidents are eliminated deterministically, comprise the basis of nuclear power production for a historically long period of time (thousands of years), are preferable from the standpoint of the low probability of serious accidents. At the same time, the inherently-safe reactors still require final design adjustments and testing. On this level, the first variant, based on existing types of reactors but with reactor safety increased up to the maximum possible level, is preferred. The third variant, where a long period of time (up to 30 years) is set aside for perfecting and checking the inherently-safe reactors, which then become the mainstay for nuclear power production, is a compromise.

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TABLE 1. Parameters of Thermal- and Fast-Neutron Reactors

Reactor	IKV (Pu)	IKV (^{233}U)	KV	B, GW(t)·yr/tons of heavy atoms	M_f per cycle, tons/GW(e)			
					Pu from fast-neutron reactors	Pu from thermal-neutron reactors	Pu from weapons	Enriched or weapons U
BN-A	0	0,3	—	0,219	9,52	9,18	8,51	19
BN-B	0,3	0	—	0,219	9,52	9,18	8,51	19
BN-C	0	0	—	0,219	9,52	9,18	8,51	19
VVÉR-A	—	—	0,7	0,126	—	—	—	3,83
VVÉR-B	—	—	0,8	0,126	—	—	—	3,83
BREST-A	0,05	0,05	—	0,219	9,47	9,13	8,47	18,9

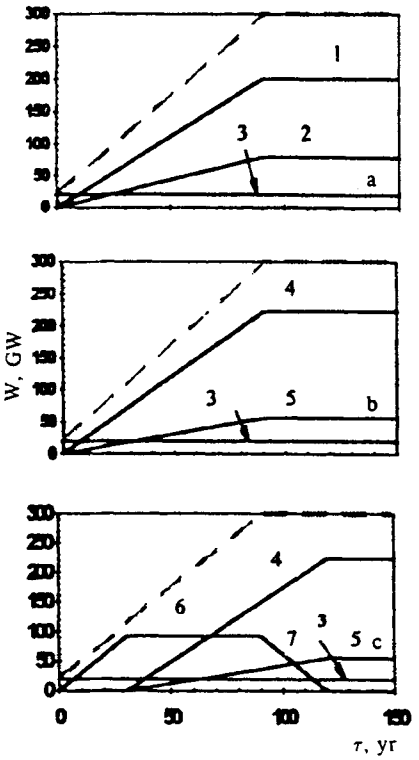

Fig. 1. Growth of the electrical capacity of nuclear power (- - -) with modern-type reactors (3) according to the variant YaÉ-1 (a) with BN-A (1) and VVÉR-A (2), YaÉ-2 (b) with BREST-A (4) and VVÉR-B (5), YaÉ-3 (c) with BN-B (6) (first stage — YaÉ-31), BN-B (6), BREST-A (4), VVÉR-B (5) second stage — YaÉ-32), BN-C (7), BREST (4), VVÉR-B (5) (third stage — YaÉ-3).

It is obvious that the safety of thermal reactors, which are introduced in order to solve problems which are not inherent to fast-neutron reactors (supplying heat, high-temperature power production, and others), must also be improved as much as possible.

Parameters of the Nuclear-Power Development. The main parameters of the improved reactors adopted at the development stage of nuclear power are presented in Table 1. Here B is the relative burnup of fuel and M_f is the mass of fissioning nuclei in the fuel cycle. The reactor modifications are designated as A, B, and C. The breeding indicators give the excess (IKV) or total (KV) number of fissioning nuclei produced, on the average, in one fission event.

TABLE 2. Indicators of Production and Burnup of Fissioning Material, Tons/(GW(e)·yr)

Reactor	Production F × KV	Burnup F	Loss F(1 + α)	Shortage F(1 + α - KV)	Excess production F × IKV
BN-A. -B	0,983	0,955	0,983	0	0,286
BN-B	0,983	0,955	0,983	0	0
VVÉR-A	0,838	1,197	1,57	0,732	0
VVÉR-B	0,942	1,178	1,297	0,355	0
BREST-A	0,92	0,894	0,92	0	0,0883

The loading per cycle takes account of a three-year external fuel cycle and the coefficient of relative value of the nuclides and their mixtures as determined in the thermal or fast neutron spectrum with respect to ^{235}U or ^{239}Pu.

The BN-800 reactor with a thorium (BN-A) and uranium (BN—B) side breeding zone is used in the first variant of development (YaÉ-1) and at the first and second stages of the third variant (YaÉ-31 and 32) as well as without the side breeding zone (BN-C) — at the third stage of the third variant (YaÉ-33). The VVÉR-1000 reactor operating on uranium–plutonium fuel with replenishment with 30% ^{233}U from BN-A (VVÉR-A) is used in YaÉ-1 and thorium-uranium fuel with replenishment with 20% ^{233}U and plutonium from the BREST-A reactors (VVÉR-B) is used in YaÉ-2 and YaÉ-32 and -33. The BREST-300 reactors with an end thorium breeding zone (BREST-A) are used in YaÉ-2, -32, and -33. The production or shortage indicators of fissioning materials are presented in Table 2.

In the self-supply regime the ratio of the electrical capacity of fast- and thermal-neutron reactors is determined by the ratio of the shortage of fissioning material in the thermal reactors to the mass of the excess production in fast reactors.

Some indicators of the different variants of nuclear power development are presented in Table 3. It is assumed that the development of nuclear power starts in 2010 (zero time) and continues for 90-120 years.

The initial load of thermal-neutron reactors consists of some of the uranium reserves for weapons (680 tons) and power production (50 tons); the initial load of fast-neutron reactors consists of the uranium left over from weapons and power production, plutonium from weapons (100 tons) and power production (50 tons), and enriched uranium. It is presumed that the uranium and plutonium for power production will be obtained in the future from reprocessing VVÉR-1000 fuel. The remaining depleted uranium will serve as reactor fuel.

The development of large-scale nuclear power requires a substantial increase in the industrial fuel-reprocessing capacity — up to 3.8-5.5 thousand tons per year, mining of natural uranium (6.5-7.2 thousand tons/yr) and thorium (150-580 tons/yr), fabrication of fuel assemblies, and so on. It is assumed that the capacity of the building industry makes it possible to introduce 3.1 GW(e) per year for the entire development period.

Different dumps and burial sites must be created, including for long-term (200 years) controllable storage of long-lived high-level wastes, and sites for their final burial. It is necessary to develop and perfect a technology of chemical reprocessing of fuel that addresses the problems of utilization and transmutation of some nuclides.

Handling of Long-Lived High-Level Wastes at the Development Stage of Nuclear Power. YaÉ-1. A closed fuel cycle with recycling of the principal actinides — uranium, plutonium, and thorium — in reactors with an initial thorium load is mandatory for nuclear power in the self-supply regime. It is desirable to eliminate from the wastes most of the Am, Cm, Np and possibly Cf or Pa from the standpoint of reducing these wastes to a minimum and approaching radiation equivalence of the wastes produced and the extracted uranium and thorium. This can be attained by burning them in power reactors, better fast reactors or in special transmutational reactors. There is a possibility of dumping these wastes in outer space or burying them for a long period of time [4]. The latter measure will make it possible to reduce for a substantial period of time (100-150 yr) the activity of actinides at a burial site in an open nuclear fuel cycle below their activity in the core during cycling in thermal-neutron reactors (Fig. 2a). This is due to the substantial increase in activity during cycling and to the approach to an equilibrium concentration of nuclides in a thermal neutron spectrum. There is a great deal of plutonium in the initial load of a fast reactor, and the activity of the actinides at a burial site exceeds from the outset the activity in the core (Fig. 2b). Since fast reactors are the foundation of nuclear power in the self-supply regime and it may be necessary to operate such reactors for much longer than 150 years, the transmutation variant is preferred over the storage variant.

The main actinides circulate during the development stage YaÉ-1 — uranium, plutonium, and thorium; Am, Np, and Cm are segregated and stored for transferral into a transmutation reactor. From the wastes 0.01% U, Pu, 0.1% Np, Am, Cm,

TABLE 3. Electrical Power and Mass of the Initial Load at the End of the Period of Development of Nuclear Power

Indicator	YaÉ-1 VVÉR-A. BN-A	YaÉ-2 VVÉR-B. BREST-A	YaÉ-31 BN-B	YaÉ-32 VVÉR-B. BREST-A BN-B	YaÉ-33 VVÉR-B. BREST-A. BN-B
Development interval, yr	0—90	0—90	0—30	30—90	90—120
$\mathcal{W}_{T.P}$, GW	78,7	55,8	0	37,2	55,8
$W_{F\,R.}$, GW	201,3	224,2	0	149,5	224,2
$W_{BN\text{-}B}$, GW	—	—	93,3	93,3	0
W_{post}, GW	20	20	20	20	20
$W_{Ya\text{-}É}$, GW	300	300	113,3	300	300
$M_{T\,P}$, tons U_{weap}	301,4	213,7	0	259**	129,5***
M_{res}, tons ^{235}U*	428,6	516,3	730	1387**	707,6***
$W_{F\,R}$ on ^{235}U, GW	22,5	27,3	38,4	—	—
$W_{F\,R}$ on Pu_{weap}, GW	11,7	11,8	11,7	—	—
$W_{F\,R}$ on Pu_{pow}, GW	5,4	5,5	5,4	—	—
$W_{F\,R}$ on Pu_{enrich}, GW	161,6	179,6	37,7**	3	—
$M_{U_{nat}}$, tons	$5{,}88\cdot10^5$	$6{,}48\cdot10^5$	0	$1{,}09\cdot10^4$	0
$M_{U_{depl}}$, tons	$5{,}81\cdot10^5$	$6{,}34\cdot10^5$	0	$1{,}06\cdot10^4$	0
M_{Th}, tons	$1{,}02\cdot10^4$	$9{,}78\cdot10^3$	0	$6{,}52\cdot10^3$	$3{,}26\cdot10^3$

*Uranium from weapons and power production.

**Excess plutonium from BN-B

***Plutonium from final loadings of BN-B.

and Cf, and 100% Th, Pa, and Bk are directed to a transmutation reactor. We note that cycling of californium is important in the thermal spectrum; cycling of protactinium becomes necessary when thorium is present in the initial load.

Reactors with cycling of metallic fuel with a fast spectrum (BTsMT) can be used as special reactors for transmutation. Fuel in the form of a melt of intermetallides in aluminum with a volume fraction of the aluminum equal to 0.93-0.94 and melting temperature 700-800°C is being studied. The phase diagrams of thorium, uranium, and plutonium [5] are similar and, in the case of almost pure aluminum, the systems Np, Am, and Cm—Al should have similar properties. On account of the fast circulation of the fuel going through phases of melting, granulation, and dispersed heat exchange with the secondary coolant, for example, the salt $NaBf_4 \cdot NaF$ with T_{melt} = 385°C [6] or sodium and separation from sodium in a cyclonic separator placed above the core, the intermetallides do not precipitate. Such reactors operate in the regime of almost 100% burnup of the additionally added fuel with rough removal of the fission products and with no reactivity excess for burnup; this increases their nuclear safety.

They can be regarded as alternatives to burnup of Np, Am, and Cm in fast power reactors and can be introduced at the operational stage of nuclear power. However, there is better justification for introducing homogeneous transmutation reactors at the stage of final decommissioning of all reactors after prolonged operation, when it becomes necessary to burn up a large mass of plutonium.

The equivalent activity of the actinide wastes in YaÉ-1 accompanying the use of special transmutation reactors is presented in Table 4. Their content in the wastes is taken in BTsMT to be equal to 0.01% of the loaded mass of Np, Am, and Cm.

As a result of transmutation, the equivalent VVÉR-1000 actinide activity drops to a level 1000 times lower than the present nuclear fuel cycle [1].

During the reprocessing of VVÉR-A and BN-A fuel, strontium and cesium, which are present in the wastes in fractions equal to 0.1 and 0.15, respectively, and technetium and iodine, whose fraction in the wastes is equal to 0.15, are segregated

TABLE 4. Equivalent Activity of Actinide Wastes in YaÉ-1, Ci/ton of Fission Products

Holding time, Yr	BN-A-800	VVÉR-A-1000	BTsMT	Average over YaÉ-1
1	118.4	409.2	18.5	205
100	77.4	41.5	10.3	64.1
200	66.3	27.6	8.7	52.5
400	51	21.3	6.5	40.4
10^3	25.9	14.1	2.9	22.2
10^4	6.6	14.9	0.36	9
10^5	1.8	15.8	0.25	6.1
10^6	2.7	2.7	0.53	2.6
10^7	0.21	0.2	0.03	0.2

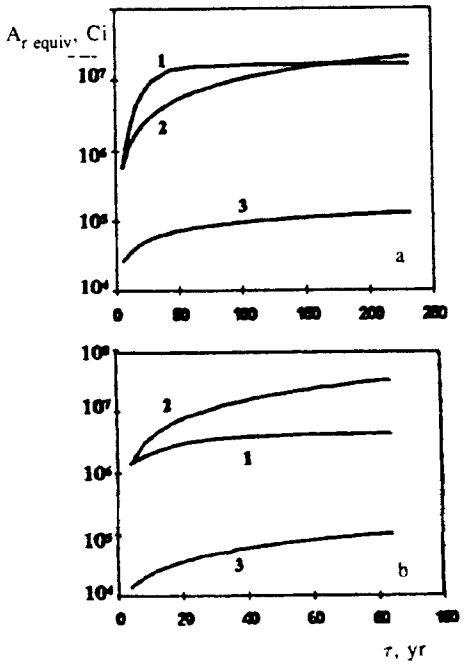

Fig. 2. Total equivalent activity of actinides produced in VVÉR-1000 (a) and BN-800 (b) reactors operating in a closed transmutation regime — core (1), wastes (3), or open fuel cycle (2).

for transmutation. Technetium and iodine are fed into BN-A. Technetium is transmuted by adding in the form TcO_2 to the BN-A fuel, and iodine in a porous-nickel matrix is transmuted in the end reflector of BN-A.

YaÉ-2. Np, Am, Cm, Tc, and I, segregated from the VVÉR-B and BREST-A fuel, enter the fuel (Np, Am, Cm) and the side reflector (technetium, iodine) in BREST-A for transmutation. The wastes acquire 0.01% uranium, plutonium, and thorium (from VVÉR), 0.1% Np, Am, Cm, and californium, and 100% protactinium, berkelium, and thorium (from the BREST core).

The large increase in the activity of VVÉR actinide wastes with holding periods of ≥ 10 yr as compared with YaÉ-1 is due to the thorium nuclear fuel cycle with 100% of the protactinium entering the wastes. The activity of the VVÉR-B actinide wastes will exceed the initial activity for a period $\sim 2 \cdot 10^4$ years; this is due to the decay of ^{232}U and ^{231}Pa and their conversion into the shorter-lived ^{228}Th and ^{227}Ac (Fig. 3, Table 5). This situation can be partially corrected by introducing protactinium recycling, but the subsequent increase in activity with a holding period of $\sim 10^5$ yr is characteristic of the thorium nuclear fuel cycle and is due to the decay of $^{233,234}U$.

TABLE 5. Equivalent Activity of Actinide Wastes in YaÉ-2, Ci/ton of Fission Products

Time, yr	BREST-A-300 (core)	VVÉR-B-1000 (Th—U NFC)	Average over YaÉ-2
1	457	348	431
100	177	460	246
200	147	436	217
400	118	418	191
10^3	78.8	412	159
10^4	44.9	376	125
10^5	11.2	186	54
10^6	5.4	261	67.2
10^7	0.33	14.4	3.7

TABLE 6. Equivalent Activity of Wastes (Ci), Mass and Activity of Uranium per ton of Fission Products

τ, yr	Development variant		
	1	2	3
100	$7.6 \cdot 10^4$	$7.45 \cdot 10^4$	$7.18 \cdot 10^4$
200	$7.81 \cdot 10^3$	$7.75 \cdot 10^3$	$7.52 \cdot 10^3$
400	629	586	586
1000	65.6	367	367
M_U, tons	56.4	65.4	6.84
A_U, Ci	265	307	32.1

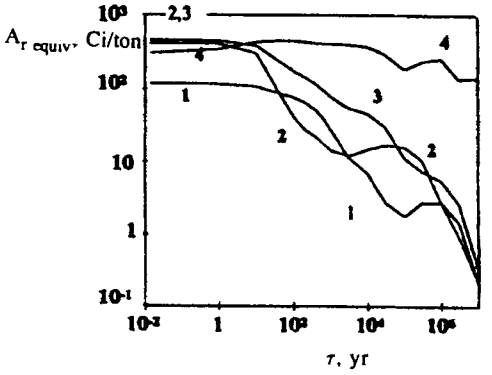

Fig. 3. Radiation-equivalent activity of actinide wastes: 1) BN-A-800, YaÉ-1; 2) VVÉR-A-1000, YaÉ-1; 3) BREST-B-1000, YaÉ-2 as a function of the holding time; 4) VVÉR-B-1000, YaÉ-2.

YaÉ-3. Intrinsic Np, Am, Cm, and Tc in the fuel and iodine in the end reflector are transmuted in BN-B (YaÉ-31, -32). The BREST-A reactors (YaÉ-32, -33) transmute Np, Am, Cm, Tc, and I, both their own nuclides and from VVÉR-B. Strontium (90%) and cesium (85%) are segregated from the fuel of all reactors for useful applications. The remaining long-lived high-level wastes are allowed to stand for a long period of time before being buried. At the third stage the uranium side breeding zones of the BN-C reactors being produced are used as replenishment fuel for BREST-A and technetium and iodine can be transmuted in their side reflector.

Activity of Long-Lived High-Level Wastes. Table 6 gives an estimate of the equivalent activity of the wastes and the activity of natural uranium for three variants of nuclear power development for an acceptable controlled holding time (200-400 yr).

For the first two variants, equivalence is achieved with a holding period of 300 yr, taking account of the 10 to 15 times lower average rate of migration of the elements of the wastes as compared with uranium. For the third variant, this time period

increases to 1000 yr because of the small amount of uranium used. To achieve equivalence with a shorter holding period, additional measures must be taken to decrease the activity of the wastes, for example, by introducing recycling of the constructional materials, decreasing the fraction of strontium in the wastes from 10 to 0.1%, or dumping some of the high-level wastes in outer space. The migration factor can also be increased up to 60 by decreasing the iodine fraction in the wastes to 0.1% and increasing the contribution of actinides to the activity of the wastes. Iodine migrates on the average along different rocks (tuff, salt, granite, basalt) five times more rapidly than uranium [7, 8]; this makes the migration rate of wastes much higher than that of natural uranium. This stimulates transmutation of iodine with a low fraction of iodine in the wastes.

In summary, a large-scale nuclear power with a capacity of 300 GW(e) can be developed in Russia over a period of 90 years. The development of nuclear power with fast- and thermal-neutron reactors in a regime of self-supply with fissioning materials requires that all reserves of natural uranium within the territory of the country (600 thousand tons) be mined for the initial loading of the fast-neutron reactors; this requires a substantial increase in the capacity of the uranium mining industry (the first two variants of the development of nuclear power). The initial development of the system of only fast reactors (third variant of development) permits avoiding almost completely the mining of natural uranium for the initial loading and for replenishment. This variant leaves a margin of 30 years for final development and adjustment of the naturally-safe reactors, for example, BREST. Over this time period, a test—commercial nuclear power plant, for example, with BREST-300, can be produced and tested comprehensively. It is believed that sodium-cooled reactors (BN-B and -C), which make possible the introduction of a system of BREST-type reactors with a low consumption of natural uranium, can be decommissioned by the end of the development stage YaÉ-3 and for the subsequent period of time the nuclear power industry will utilize mainly naturally-safe reactors which make possible highly safe operation over a long period of time. When some of these measures are adopted, including a high degree of separation of iodine for subsequent transmutation, the radiation and migration equivalence of the long-lived high-level wastes and natural uranium is achieved in all three variants of the development of nuclear power with an acceptable period of time ($\sim$200 yr) for controllable holding of the wastes prior to final burial.

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