NUCET NUCLEAR ENERGY AND TECHNOLOGY

Nuclear Energy and Technology 11(1): 33–41
DOI 10.3897/nucet.11.127108

Research Article

Closed nuclear fuel cycle of thermal and fast reactors with fuel self-sufficiency

Anatoly N. Shmelev[1], Vladimir A. Apse[1], Evgeny G. Kulikov[1], Gennady G. Kulikov[1], Vasily B. Glebov[1]

1 *National Research Nuclear University MEPhI (Moscow Engineering Physics Institute), 115409, Moscow, Russia*

Corresponding author: Vasily B. Glebov (vbglebov@mephi.ru)

Academic editor: Piero Ravetto ♦ **Received** 11 May 2024 ♦ **Accepted** 11 December 2024 ♦ **Published** 14 February 2025

Citation: Shmelev AN, Apse VA, Kulikov EG, Kulikov GG, Glebov VB (2025) Closed nuclear fuel cycle of thermal and fast reactors with fuel self-sufficiency. Nuclear Energy and Technology 11(1): 33–41. https://doi.org/10.3897/nucet.11.127108

Abstract

The paper presents the results obtained in numerical evaluations of a possibility to reach self-sufficiency of fissile materials in the joint system of fast and thermal reactors. These studies considered the joint system consisting of thermal light-water reactors of VVER-type and fast lead-cooled reactors of BREST-type, which are operated within the frames of the closed (Th-U-Pu) fuel cycle. It was assumed that fast reactors (FR) of BREST-type used the mixed thorium-plutonium nitride fuel, while thermal reactors (TR) of VVER-type used the mixed oxide fuel of natural uranium and ^{233}U. Uranium isotope ^{233}U was produced in the Th-fraction of FR fuel for further introduction into the fresh composition of TR fuel, while plutonium was produced in the natural uranium fraction of TR fuel for further introduction into the fresh composition of FR fuel. The numerical studies resulted in the determination of the conditions necessary to provide fuel self-sufficiency in the joint TR-FR system. The following key findings of the research may be noted:

– It is demonstrated the possibility to create the joint TR-FR system with inherent fuel self-sufficiency;

– Involvement of thorium and ^{233}U in the closed NFC of the joint TR-FR system can arrange an optimal regime for production and consumption of main FM;

– Purposeful change of thermal power and introduction of natural uranium in the fuel composition of BREST-type FR made it possible to reach fuel self-sufficiency of the joint TR-FR system;

– Application of radiogenic lead instead of natural lead allowed us to reduce necessary values of thermal power and content of natural uranium nitride in the fuel of BREST-type FR.

Keywords

fast BREST-type reactors, fuel self-sufficiency, thermal VVER-type reactors

Introduction

The national program of the Russian Federation on innovative development of the nuclear power industry foresees creating the two-component nuclear power system (NPS) based on thermal light-water VVER-type reactors and fast reactors. The operational start-up of sodium-cooled fast reactor BN-800 in 2015 and the use of mixed oxide (MOX) uranium-plutonium fuel in this reactor presented the practical beginning for real creation of the two-component NPS.

Main advantages of the two-component NPS:

- Utilization of depleted and natural uranium as a feeding fuel of nuclear power reactors.
- Utilization of plutonium stockpiles.

- Reprocessing of spent nuclear fuel (SNF).

Many publications (for example, Andrianova et al. 2008; Dekusar et al. 2010; Kagramanyan et al. 2014; Alekseev et al. 2016; Petrov et al. 2019; Zrodnikov et al. 2021) have analyzed various models of the closed nuclear fuel cycles. Some systems of thermal and fast nuclear power reactors were evaluated from the standpoint of a possibility to close the nuclear fuel cycle (NFC) based on application of MOX-fuel with plutonium recycling.

For example, the publication (Alekseev et al. 2016) declared that the problems related to SNF reprocessing, radioactive waste management, and limitation of nuclear fuel resources might cardinally be solved by creating the closed-cycle NPS consisting of thermal and fast nuclear power reactors. The real foundation for successful formation of the two-component NPS in the Russian Federation is the contemporary Russian nuclear power with thermal VVER-type reactors and the sufficiently broad technological experience gained in operation of sodium-cooled fast reactors (BN-type FR).

Two basic scenarios of the NPS development in the Russian Federation were analyzed in the publication (Alekseev et al. 2016):

Scenario A: Operation of VVER-type TR within the frames of the open NFC. Total power of the system will increase by 2035 in accordance with the RF Target Roadmap.

Scenario B: Operation of VVER-type TR and BN-type FR within the frames of the closed NFC with plutonium recycling.

The following conclusions were derived from the obtained results:

- The optimal ratio between VVER-type TR and BN-type FR in the joint NPS from the viewpoint of fuel balance and fuel quality appeared to be 2:1.
- Highly efficient solutions to the NPS problems were found for the following aspects: specific consumption rate of natural uranium; amount of radioactive waste to be ultimately disposed of; reliable proliferation resistance; innovative potentials of the NPS.
- The problem related to the degradation of plutonium isotope composition in VVER-type TR may be resolved by the partial introduction of MOX-fuel assemblies together with traditional UOX-fuel assemblies into the fresh fuel loading. The nuclear fuel cycle is partially closed, mainly for BN-type FR.

The publication (Petrov et al. 2019) considers the operation of the two-component NPS within the frames of the closed NFC. BN-type FR is used to produce plutonium with isotope composition the best suitable to fabricate MOX-fuel for VVER-type TR. Main attention is given to BN-type FR because the industrially mature, safe, and competitive technology for FR design, construction, and operation is currently available in the Russian Federation. VVER-type TR applies MOX fuel instead of traditional UOX fuel. Plutonium extracted from spent VVER fuel is used in MOX fuel fabrication. Layout of the closed NFC is shown in Fig. 1.

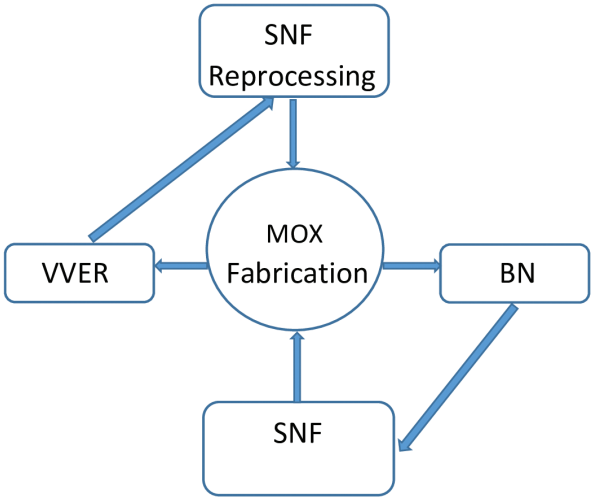

Figure 1. Layout of mutual fuel exchange in TR-FR system.

The scheme of the closed NFC with application of the unified MOX fuel in both types of nuclear power reactors was proposed in the publication (Petrov et al. 2019). Such an approach is able to significantly diminish the expenses for MOX-fuel fabrication. However, a possibility for multiple recycling of the unified MOX-fuel composition in VVER-type TR remained unknown.

The results obtained in detailed analysis of various development scenarios of the two-component NPS with the closed NFC are presented in the publication (Zrodnikov et al. 2021) with proper accounting for potential export capabilities of Russian nuclear technologies. It is assumed that the NFC of the two-component NPS is closed by plutonium recycling only. Spent fuel assemblies from both reactor types are reprocessed; the extracted plutonium is used in the fabrication of fresh fuel assemblies for BN-type FR and, if necessary, in the fabrication of fresh MOX-fuel assemblies for VVER-type TR. The schemes of fuel supplies satisfied the following conditions: the two-component NPS must not consume plutonium from external stockpiles and accumulate excessive plutonium amounts. The mass balance between the loaded and discharged plutonium amounts is reached by the properly chosen ratio between the numbers of TR and FR units as well as by operational parameters of both reactor types.

The maximal achievable values for total established power of the two-component NPS were assessed in the publication (Zrodnikov et al. 2021) as functions of the breeding ratio in BN-type FR and limitation on natural uranium resources. The possibility was shown to change the NPS power within a certain range under the following conditions: spent fuel must not be accumulated and uranium consumption must be shortened. This capability is an important criterion for the selection of the two-component NPS structures and relevant technologies for the NPS development. It was shown the established power of

the two-component NPS based on operation of BN-type FR with MOX-fuel and VVER-type TR with UOX-fuel could reach 300 GWe if the breeding ratio equals 1.4 and if the external part of the NFC is 4-years long.

The two-component NPS models were analyzed in the publications (Andrianova et al. 2008; Dekusar et al. 2010; Kagramanyan et al. 2014; Alekseev et al. 2016; Petrov et al. 2019; Zrodnikov et al. 2021) from the viewpoint of potential NFC closure and plutonium recycling as a main fissile material (FM) in MOX-fuel compositions. However, if NFC became closed, then operation of thermal reactors (for instance, VVER-type TR) would meet some principal difficulties. The main difficulty is related to the rapid degradation of plutonium isotope composition caused by its shifting towards the heavier plutonium isotopes. The deterioration results either in the lower number of plutonium recycling in VVER-type TR (no more than twice-fold recycling) or in the necessity to enhance substantially the breeding plutonium properties in fast reactors.

The following ways may be used to increase multiplicity of fuel recycling in the two-component NPS, namely full exclusion of Pu-bearing materials in fuel loading of thermal reactors and involvement of the fissile materials allowing their multiple recycling (for example, ^{233}U from FR).

The problems related to the practical involvement of ^{233}U-bearing materials into fuel compositions of thermal reactors were analyzed in a lot of publications. For example, the paper (Ponomarev-Stepnoy et al. 1998) presents the project of light-water reactor VVER-T with thorium-bearing materials. The VVER-T project presumes the use of heterogeneous fuel assemblies with mixed uranium-thorium fuel in the blanket zone and mixed uranium-zirconium fuel in the driver zone of VVER-1000 instead of traditional UOX-fuel assemblies. The VVER-T project can provide the following positive results:

- About 30% of thermal energy may be produced by ^{233}U without recycling.
- consumption rate of natural uranium may be reduced by 20%;
- The risk of FM proliferation may be reduced by keeping the content of fissile uranium isotopes at a level below 20% in all the reactor operation modes;
- The production rate of minor actinides may be significantly decreased.

The present paper considers the joint TR-FR system based on the unification of U-Pu and U-Th fuel cycles. The joint TR-FR system is the closed system for ^{233}U and plutonium with internal FM reproduction. The proposed TR-FR system will operate under the following conditions:

- Fuel self-sufficiency with external feeding by natural uranium and thorium only;
- Multiple fuel recycling in the partial refueling regime;
- SNF reprocessing for the further fuel feeding by "fresh" ^{233}U and plutonium.

Recycle of fissile materials in the joint TR-FR system

Unification of U-Pu and U-Th fuel cycles within the frames of the joint TR-FR system opens an opportunity for optimal disposition, production, and consumption of different FM between thermal and fast reactors. In particular, the combination of two fuel cycles made it possible to load the system's reactors with different FM. For example, plutonium utilization in VVER-type TR seems unreasonable because of the rapid degradation of neutron-multiplying properties. Simultaneously, application of ^{233}U as a main fissile isotope in thermal reactors can provide a remarkable profit in neutron economy because, within thermal energy range, ^{233}U is able to produce about 0.2 "additional" neutrons as compared with ^{235}U and ^{239}Pu (Subbotin 2007). This allows us to regard ^{233}U as a promising FM for VVER-type TR. In addition, high FR capabilities in FM breeding (including intense ^{233}U production from fertile thorium) make it reasonable to arrange ^{233}U production in BREST-type FR.

If plutonium is produced and consumed in BREST-type FR, then ^{233}U cannot be accumulated in sufficient amounts for loading into VVER-type TR. That is why plutonium should be produced in VVER-type TR. The missing mass of plutonium from VVER-type TR can be compensated by the self-produced plutonium in BREST-type FR.

Fig. 2 shows a layout of the exchange of fissile materials between the reactors of the TR-FR system. ^{233}U from BREST-type FR is used as a main FM in VVER-type TR while plutonium from VVER-type TR is used as a main FM in BREST-type FR. With this FM exchange scheme, regular feeding of these reactors with "fresh" ^{233}U and plutonium is possible within the framework of a single process of radiochemical processing of irradiated materials and no zone allocation for special FM production.

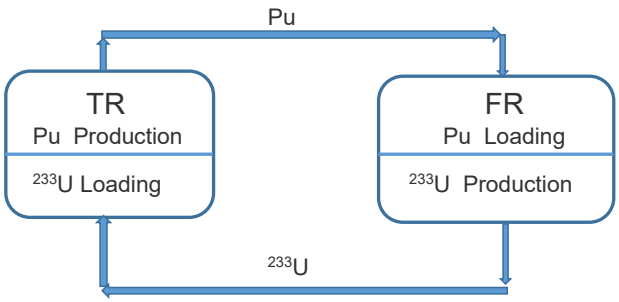

Figure 2. Layout of FM production and consumption in the joint TR-FR system.

An important specific feature of FM recycling in the joint TR-FR system is reduced degradation of U and Pu neutron-multiplying properties during multiple irradiation cycles. Reduced degradation is achieved due to:

- regular feeding of the reactors with "fresh" ^{233}U and plutonium;

- irradiation of plutonium in FR;
- high-grade uranium sent from FR to TR.

Specificity of neutron reactions in fast reactors with plutonium loading

If TR-produced plutonium is used in FR as a main fissile material, then neutron irradiation can cause some changes in plutonium isotope composition. Energy-dependent microcross-sections of fission reaction and radiative neutron capture by ^{239}Pu are presented in Fig. 3 (Janis Web 2024). The following conclusions can be derived from these dependencies.

Within the typical neutron energy range of fast reactors (0.01÷10 MeV), fission reaction dominates over radiative neutron capture by odd plutonium isotopes ^{239}Pu and ^{241}Pu. Within this energy range, microcross-sections of fission reactions for even plutonium isotopes ^{240}Pu and ^{242}Pu are significantly lower as compared with analogous values for odd plutonium isotopes, while microcross-sections of radiative neutron capture are comparable with those for odd plutonium isotopes (Janis Web 2024). Radiative neutron capture either improves plutonium isotope composition by transformation of ^{240}Pu to ^{241}Pu or converts plutonium into another chemical element (transformation of ^{242}Pu to ^{243}Am).

In general, the specificity mentioned above allowed us to expect moderate changes in mean neutron-multiplying properties of plutonium during multiple irradiation cycles in BREST-type FR.

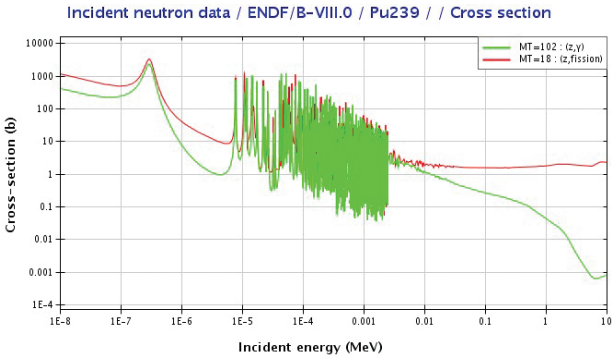

Figure 3. Energy-dependent microcross-sections of fission reaction and radiative neutron capture by ^{239}Pu.

Specificity of neutron reactions in thermal reactors with uranium loading

In thermal reactors, ^{233}U is a well-fissile isotope ($\sigma_f \geq 100$ barns), and fission reaction dominates over radiative capture reaction (Fig. 4). Besides, micro cross-sections of radiative neutron capture by the next uranium isotope ^{234}U are substantially higher (by three orders of magnitude) as compared with micro cross-sections of its fission reaction (Janis Web 2024). So large micro-cross-sections of neutron capture by ^{234}U can lead to intense conversion of ^{234}U to ^{235}U. Thus, it can improve uranium isotope composition under irradiation of ^{233}U in thermal reactors.

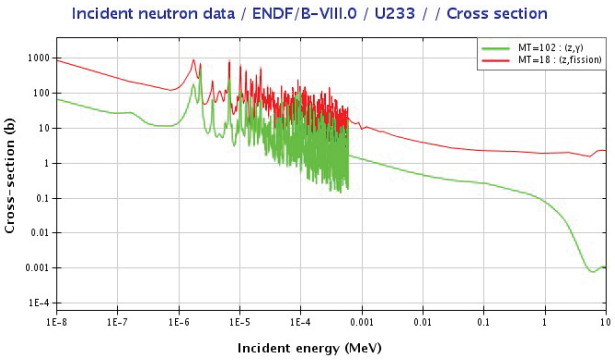

Figure 4. Energy-dependent microcross-sections of fission reaction and radiative neutron capture by ^{233}U.

The specificity mentioned above allowed us to expect moderate changes in mean neutron-multiplying properties of the uranium during successive irradiation cycles in VVER-type TR.

Use of FM fractions in reactors under operation in a partial refueling regime

The scheme of FM production and consumption in the joint TR-FR system presumes that plutonium extracted from spent fuel of VVER-type TR is used to feed BREST-type FR while uranium extracted from spent fuel of BREST-type FR is used to feed VVER-type TR (Fig. 2). It should be noted here that this FM exchange scheme implies a common operating cycle of the TR and FR reactors, which is usually implemented within the framework of partial fuel reloadings.

In general, if reactors of the joint TR-FR system are operated in a partial refueling regime with multiplicity of refueling K, then the average plutonium composition of FR loading in the stationary operation mode can be determined as follows: $\overrightarrow{\rho_L}(Pu) = \left(\sum_{i=1}^{K} \overrightarrow{\rho_i}(Pu)\right)/K$ where $\overrightarrow{\rho_1}(Pu) = \overrightarrow{\rho_F}(Pu)$ – isotope composition of plutonium extracted from spent fuel of VVER-type TR and used to feed BREST-type FR. Parameters of the partial refueling regime are mainly defined by the requirement to reach a certain level of the reactor supercriticality after refueling.

Similarly, the average uranium composition of TR loading in the stationary operation mode can be determined as follows: $\overrightarrow{\rho_L}(U) = \left(\sum_{i=1}^{K} \overrightarrow{\rho_i}(U)\right)/K$, where $\overrightarrow{\rho_1}(U) = \overrightarrow{\rho_F}(U)$ – the isotope composition of uranium extracted from spent fuel of BREST-type FR and used to feed VVER-type TR.

The listed above considerations about specificity of neutron reactions with plutonium isotopes in fast reactors and neutron reactions with uranium isotopes in thermal reactors allowed us to apply the following simplified model of the NPS operation in a partial refueling regime: Average plutonium composition of FR loading differs slightly from isotope composition of plutonium extracted

from spent fuel of VVER-type TR and used to feed BREST-type FR, i.e., $\overrightarrow{\rho_L}(Pu) \approx \overrightarrow{\rho_1}(Pu) = \overrightarrow{\rho_F}(Pu)$. Similarly for VVER-type TR $\overrightarrow{\rho_L}(U) \approx \overrightarrow{\rho_1}(U) = \overrightarrow{\rho_F}(U)$.

The above assumptions correspond to the case of refueling multiplicity K = 1. Under these conditions, the possibility to reach fuel self-sufficiency in the joint NPS consisting of VVER-type TR on uranium dioxide fuel and BREST-type FR on thorium-plutonium nitride fuel can be assessed by numerical studies. In addition to this, it is also necessary to consider the issues of applying the partial reloading mode to the TR-FR system in a more general case. At present, the issues of organizing the system operation in the double reloading mode are studied. The plans for implementing this mode also include the influence of operating cycle duration and the exchange of reactors with multicomponent mixtures of nuclear materials.

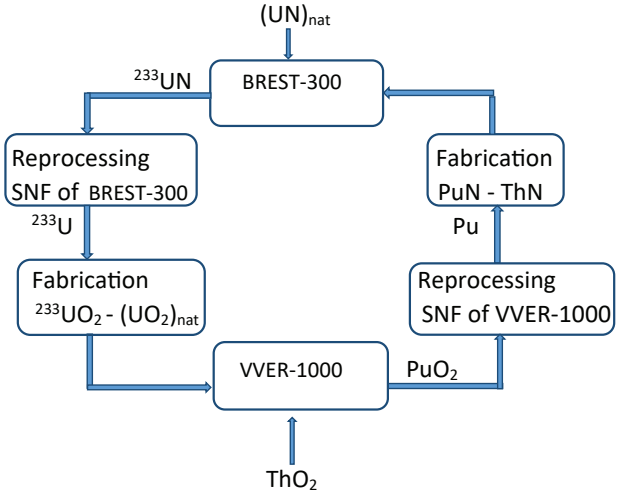

Figure 5. Layout of material flows in the joint TR-FR system.

Numerical studies of the joint TR-FR system

Mathematical model

The main purpose of neutron-physical studies was the determination of conditions for fuel self-sufficiency of light-water VVER-1000 TR and lead-cooled BREST-300 FR in the joint NPS. The following assumptions were used in calculations:

1. ^{233}U for VVER-type TR is produced in BREST-type FR fuelled with mixed thorium-plutonium nitride fuel. Nitride of natural uranium could be added to thorium-uranium nitride fuel for partial compensation of plutonium burn-up.
2. Plutonium for BREST-type FR is produced in VVER-type TR fuelled with mixed dioxide of natural uranium and dioxide of ^{233}U produced in BREST-type FR. Similarly, thorium dioxide could be added to UOX-fuel for partial compensation of ^{233}U burn-up.
3. Two variants of lead coolant were analyzed in calculations of BREST-type FR, namely natural lead and radiogenic lead with 100% content of lead isotope ^{208}Pb (extremely weak neutron absorber).
4. Fuel self-sufficiency of the joint TR-FR system with balance on the production rate and the consumption rate of ^{233}U was reached by properly changing the thermal power of BREST-type FR.
5. Fuel self-sufficiency of the joint TR-FR system with balance on the plutonium production rate and the plutonium consumption rate was reached by introducing nitride of natural uranium in the fuel composition of BREST-type FR.

Layout of material flows in the joint NPS consisting of thermal VVER-1000 reactors and fast BREST-300 reactors is presented in Fig. 5.

As is seen from the layout, fast reactor BREST-300 is fuelled with thorium nitride (Alekseev et al. 2014) in a mixture with nitride of plutonium (Fedorov et al. 2021) produced in thermal reactor VVER-1000. The major mission of the BREST-300 reactor is to produce ^{233}U for further use as a main FM in the VVER-1000 reactor. Thermal reactor VVER-1000 is fuelled with dioxide of natural uranium in a mixture with dioxide of ^{233}U produced in fast reactor BREST-300. The major mission of the VVER-1000 reactor is to produce plutonium for further use as a main FM in the BREST-300 reactor. Nitride of natural uranium can be used as a possible admixture to the mixed thorium-plutonium nitride fuel of the BREST-300 reactor for partial compensation of plutonium burn-up. Similarly, thorium dioxide can be used as a possible admixture to the mixed UOX fuel of the VVER-1000 reactor for partial compensation of ^{233}U burn-up.

The parameters of the thermal VVER-1000 reactor and the fast BREST-300 reactor used in neutron-physical calculations are presented in Tables 1, 2. A one-year time period was adopted as a refueling time interval. The equilibrium content of uranium-232 in the system was ~ 0.01%. The cooling time before reprocessing and manufacturing was taken to be very short.

Table 1. Parameters of thermal reactor VVER-1000 (Andrushechko et al. 2010)

Parameter	
Thermal power	3200 MW
Radius of the reactor core	175 cm
Height of the reactor core	353 cm
Diameter of fuel pellet	7.57 mm
Diameter of central hole	1.40 mm
Thickness of fuel-cladding gap	0.075 mm
Thickness of cladding	0.65 mm
Pitch of fuel lattice	12.75 mm
Fuel density	10.5 g/cm^3
Material of cladding	99% Zr, 1% Nb
Density of material of cladding	6.5 g/cm^3
Water density	0.73 g/cm^3

Table 2. Parameters of fast reactor BREST-300 (Borisov et al. 2000)

Parameter	
Thermal power	700 MW
Radius of the reactor core	115 cm
Height of the reactor core	110 cm
Diameter of fuel pellet in small-diameter-zone (SDZ)	7.7 mm
Diameter of fuel pellet in middle-diameter-zone (MDZ)	8.2 mm
Diameter of fuel pellet in large-diameter-zone (LDZ)	9.0 mm
Thickness of fuel-cladding gap	0.2 mm
Thickness of cladding	0.5 mm
Pitch of fuel lattice	13.6 mm
Density of thorium nitride	11.5 g/cm^3
Density of plutonium nitride	14.32 g/cm^3
Density of uranium nitride	14.32 g/cm^3
Density of lead	10.47 g/cm^3
Material of cladding	stainless steel EP-823
Density of material of cladding	7.9 g/cm^3
Numbers of fuel assemblies in SDZ, MDZ and LDZ	57, 72, 56

Initial fuel inventories and effective neutron multiplication factors are presented in Table 3.

Table 3. Initial inventory and neutron multiplication factor

BREST-type fast reactor	ThN, ton	PuN, ton	Initial K_{ef}
	14.8	2.5	1.0
VVER-type thermal reactor	$(UO_2)_{nat}$, ton	$(^{233}UO_2)$, ton	Initial K_{ef}
	95.4	1.6	1.40

Two variants of coolant were used in calculations, namely natural lead and radiogenic lead (100% ^{208}Pb).

Neutron-physical calculations were carried out with application of the computer code TIME26 (Kuzmin et al. 2015) for numerical analysis of cylindrical models of nuclear power reactors in 26-group diffusion approximation. Microcross-sections were taken from the evaluated nuclear data library ABBN-78 (Abagyan et al. 1981). These data were processed by the auxiliary computer code ARAMACO-S1 to prepare the blocked microcross-sections for each zone of the reactor core. The applicability of the BNAB-78 library to the calculation of thermal reactors is considered, for example, in the articles (Abagyaand Yudkevich 1977; Abagyan et al. 1989).

The computer code TIME26 was also used to determine time-dependent evolution of fuel isotope compositions in both reactor types by solving burn-up equations for the chain of isotopic transformations shown in Fig. 6.

Algorithm of searching for fuel self-sufficiency conditions in the joint TR-FR system

The iterative searching for fuel self-sufficiency conditions in the joint TR-FR system was begun from neutron-physical analysis of the thermal VVER-1000 reactor. In initial iteration, it was assumed that uranium produced in the fast BREST-300 reactor for further use as the main FM of the thermal VVER-1000 reactor consisted of ^{233}U only. The next step included the determination of such a fraction of uranium (a mixture of natural uranium and ^{233}U) that provides the reactivity margin high enough for one-year-long operation of VVER-1000. Then, the burn-up equations were solved to determine the isotope composition of VVER-1000 fuel at the end of one-year operation, mainly mass and isotope composition of plutonium for further introduction into fresh fuel of BREST-300, as shown in Fig. 5. The next step of the iterative process included neutron-physical analysis of BREST-300, with the main goal of determining such a fraction of plutonium nitride produced in VVER-1000 in a mixture with thorium nitride that provides the critical state of BREST-300. The burn-up equations were again solved to determine isotope composition of BREST-300 fuel at the end of one-year operation, mainly mass and isotope composition of uranium (^{233}U-^{232}U mixture) for further introduction into fresh fuel of VVER-1000. After this computation, the iterative process came back to the neutron-physical analysis of VVER-1000 with the new isotope composition of uranium produced in BREST-300. The iterative process continued till stabilization of uranium isotope composition produced in BREST-300 and plutonium isotope composition produced in VVER-1000.

The possibility of reaching fuel self-sufficiency in the joint system of VVER-type TR and BREST-type FR may be evaluated via determination of plutonium production rate in VVER-1000 and ^{233}U production rate in BREST-300. Then, these production rates must be compared with plutonium consumption rates in BREST-300 and ^{233}U consumption rates in VVER-1000.

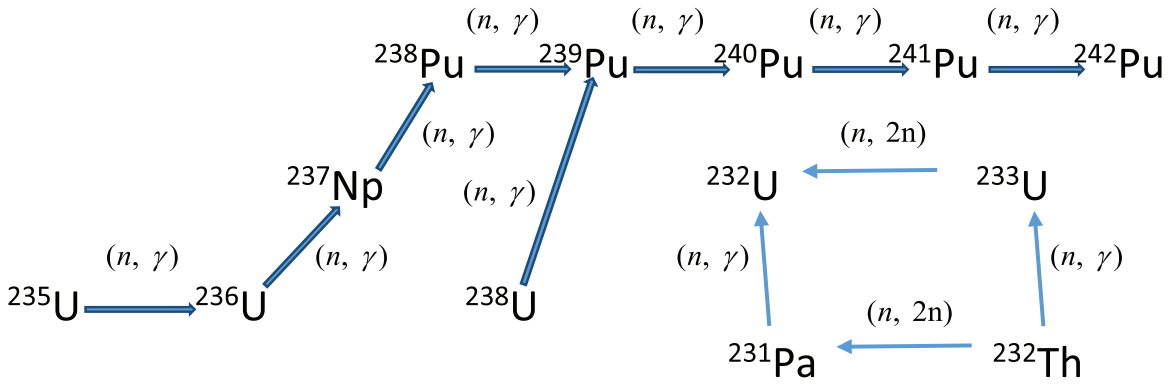

Figure 6. Chain of isotopic transformations in (Th-U-Pu)-fuel.

To satisfy VVER-type TR demands for ^{233}U, thermal power W_T (FR) of BREST-type FR must be changed in such a way:

$$W_T'(FR) = W_T(FR) \cdot M_{233}(TR) / M_{233}(FR - W_T) \quad (1)$$

where $M_{233}(TR)$ is the consumption rate of ^{233}U in TR; $M_{233}(FR\text{-}W_T)$ is the production rate of ^{233}U in FR with thermal power $W_T(FR)$.

Then the joint TR-FR system consisting of VVER-type TR and BREST-type FR with properly changed thermal power $W_T'(FR)$, or several BREST-type reactors with total thermal power $W_T'(FR)$, is balanced on ^{233}U production rate in TR and ^{233}U consumption rate in FR.

However, the question arises about the balance between plutonium production rate in VVER-type TR and plutonium consumption rate in BREST-type FR with new thermal power. The changed plutonium consumption rate in BREST-type FR can be determined in such a way:

$$M_{Pu}(FR - W_T') = M_{Pu}(FR - W_T) \cdot W_T'(FR) / W_T(FR). \quad (2)$$

Then the difference between plutonium production rate in VVER-type TR and plutonium consumption rate in BREST-type FR with new thermal power, i.e., plutonium disbalance (or deficit), can be determined in such a way:

$$\Delta M_{Pu} = M_{Pu}(TR) - M_{Pu}(FR - W_T') = M_{Pu}(TR) - M_{Pu}(FR - W_T) \cdot W_T'(FR) / W_T(FR). \quad (3)$$

Neutron-physical analyses of the joint TR-FR system

The following TR-FR system was considered in the first series of calculations.

VVER-type TR: A mixture of natural uranium dioxide with dioxide of ^{233}U produced in BREST-type FR was used as UOX-type fuel. The content of ^{233}U dioxide was chosen to provide the necessary value for the initial reactivity margin. The function of the reactor in the system was to produce plutonium for further use as a main FM in BREST-type FR.

BREST-type FR: A mixture of thorium nitride with nitride of plutonium produced in VVER-type TR was used as a fuel. The content of plutonium nitride was chosen to provide initial criticality of the reactor. Nitride of natural uranium could be introduced into fuel composition as a fertile material to reduce plutonium consumption rate. The function of the reactor in the system was to produce ^{233}U for further use as a main FM in VVER-type TR. Natural lead was used as a coolant.

The results obtained in the first series of calculations on FM production and consumption rates in the joint TR-FR system are presented in Table 4.

The data presented in Table 4 allowed us to make the following conclusions. If fuel of BREST-300 does not contain nitride of natural uranium, then the joint TR-FR system may be balanced only on the rates of ^{233}U production and consumption by properly changing the thermal power of BREST-300. However, disbalance between the rates of plutonium production and consumption in the system reaches the value of 226.6 kg/year. The introduction of natural uranium nitride into mixed thorium-plutonium nitride fuel appeared to be able to reduce the disbalance drastically. If content of natural uranium nitride in fuel composition increased up to 50%, then plutonium disbalance became positive. So, these results demonstrated a possibility of principle to reach full balance between production and consumption rates of main FM (^{233}U and plutonium) in the joint TR-FR system.

The second series of calculations differed from the first one only by the application of radiogenic lead (100% ^{208}Pb) as a coolant in the fast BREST-300 reactor instead of natural lead. The results obtained in the second series of calculations on FM production and consumption rates in the system are presented in Table 5.

Comparison of data presented in Table 4 and Table 5 showed the preference of using radiogenic lead as a coolant of BREST-type FR. Full balance between production and consumption rate of ^{233}U and plutonium was reached at the lower content of natural uranium nitride in the fuel composition of BREST-type FR. The preferability of radiogenic lead in comparison with natural lead is caused by extremely weak neutron absorption by the lead isotope ^{208}Pb, a major component of radiogenic lead.

In the third series of calculations, the possibility was assessed to compensate, at least partially, the burn-up of ^{233}U produced in BREST-type FR and used in VVER-type TR as a main FM by introducing thorium dioxide into the fuel composition of VVER-type TR. The following TR-FR system was considered in the third series of calculations.

Table 4. FM production and consumption rates in the joint TR-FR system*)

$\varepsilon(UN)_{nat}$, %	VVER-1000		BREST-300		$W_T'(FR)$, MW	ΔM_{Pu}, kg/year
	$M_{Pu}(TR)$, kg/year	$M_{233}(TR)$, kg/year	$M_{Pu}(FR)$, kg/year	$M_{233}(FR)$, kg/year		
0	+ 491.3	- 717.6	- 230.2	+ 230.1	2183	- 226.6
10	+ 491.4	- 717.6	- 200.6	+ 208.1	2414	- 200.3
20	+ 491.2	- 717.6	- 170.6	+ 185.5	2708	- 168.8
30	+ 491.0	- 717.7	- 140.1	+ 162.3	3095	- 128.5
40	+ 491.0	- 717.8	- 108.9	+ 138.1	3638	- 75.0
50	+ 491.0	- 717.8	- 76.5	+ 113.0	4447	+ 5.1

*) $M_{233}(TR)$ is the consumption rate of ^{233}U in TR; WT' (FR) is the changed thermal power of FR; $\varepsilon(UN)_{nat}$ is the fraction of natural uranium nitride in the fuel of FR; $M_{Pu}(TR)$ is the plutonium production rate in TR; WT'(FR) is the properly changed thermal power of FR; ΔM_{Pu} is the plutonium disbalance between the plutonium production rate in TR and the plutonium consumption rate in FR.

Table 5. FM production and consumption rates in the joint TR-FR system

$\varepsilon(UN)_{nat}$, %	VVER-1000		BREST-300		W_T'(FR), MW	ΔM_{Pu}, kg/year
	M_{Pu}(TR), kg/year	M_{233}(TR), kg/year	M_{Pu}(FR), kg/year	M_{233}(FR), kg/year		
0	+ 491.0	- 717.7	- 227.6	+ 230.3	2181	- 218.3
10	+ 491.0	- 717.7	- 197.0	+ 208.5	2410	- 187.1
20	+ 491.0	- 717.7	- 166.5	+ 186.3	2697	- 150.4
30	+ 491.0	- 717.7	- 135.5	+ 163.2	3079	- 104.9
40	+ 491.0	- 717.6	- 103.7	+ 139.7	3596	- 41.7
45	+ 491.0	- 717.6	- 87.7	+ 127.5	3940	- 1.6
50	+ 491.0	- 717.6	- 71.2	+ 115.0	4370	+ 46.7

Table 6. FM production and consumption rates in the joint TR-FR system

$\varepsilon(ThO_2)$, %	VVER-1000		BREST-300		W_T'(FR), MW	ΔM_{Pu}, kg/year
	M_{Pu}(TR), kg/year	M_{233}(TR), Kg/year	M_{Pu}(FR), kg/year	M_{233}(FR), kg/year		
0	+ 491.3	- 717.6	- 230.2	+ 230.1	2183	- 226.6
10	+ 438.4	- 732.0	- 230.1	+ 230.5	2223	- 292.3
20	+ 396.8	- 735.9	- 230.0	+ 230.7	2233	- 336.3
30	+ 358.5	- 733.6	- 230.0	+ 230.8	2225	- 372.6
40	+ 321.2	- 727.0	- 230.0	+ 230.9	2204	- 403.0
50	+ 283.5	- 716.0	- 230.0	+ 230.9	2171	- 429.7

VVER-type TR: A mixture of natural uranium dioxide with dioxide of ^{233}U produced in BREST-type FR was used as UOX-type fuel. The content of ^{233}U dioxide was chosen to provide the necessary value for the initial reactivity margin. Thorium dioxide was introduced into fuel composition as a fertile material to reduce ^{233}U consumption rate.

BREST-type FR: A mixture of thorium nitride with nitride of plutonium produced in VVER-type TR was used as a fuel. The content of plutonium nitride was chosen to provide initial criticality of the reactor. Nitride of natural uranium was not introduced into fuel composition. Natural lead was used as a coolant.

The results obtained in the third series of calculations on FM production and consumption rates in the joint TR-FR system are presented in Table 6.

The results obtained in the third series showed inability of thorium dioxide to improve the situation with ^{233}U burn-up and plutonium production in VVER-type TR. Partial substitution of thorium dioxide for natural uranium dioxide did not result in remarkable reduction of ^{233}U burn-up rate. Simultaneously plutonium production rate was substantially decreased. So negative effect is mainly caused by weak fuel breeding properties of VVER-type TR. Supposedly, the situation may be improved by the use of light-water reactor with super-critical parameters of coolant in the joint TR-FR system. As is known, light-water reactor with super-critical parameters are characterized with relatively better fuel breeding properties as compared with VVER-type TR.

Conclusions

The paper presents the results obtained in numerical studies of the possibility to create the joint TR-FR system with inherent fuel self-sufficiency, i.e., with a mutually balanced production rate and consumption rate of ^{233}U and plutonium. The obtained results allowed us to make the following conclusions:

1. Involvement of thorium and ^{233}U in the closed NFC of the joint TR-FR system can arrange an optimal regime for production and consumption of main FM. In particular, the reactors of the system may be fed with "fresh" FM, i.e., VVER-type TR may be fed with non-recycled ^{233}U, and BREST-type FR may be fed with non-recycled plutonium.
2. Purposeful change of thermal power and introduction of natural uranium nitride into mixed thorium-plutonium nitride fuel composition of BREST-type FR made it possible to reach fuel self-sufficiency of the joint TR-FR system on production and consumption rates of ^{233}U and plutonium.
3. Application of radiogenic lead with a dominant content of ^{208}Pb instead of natural lead as a coolant of BREST-type FR resulted in remarkable improvement of fuel self-sufficiency conditions in the joint TR-FR system. Extremely low neutron absorption by radiogenic lead allowed us to reduce necessary values of thermal power and content of natural uranium nitride in mixed thorium-plutonium nitride fuel of BREST-type FR.
4. Introduction of thorium dioxide into mixed oxide fuel of VVER-type TR for partial compensation of ^{233}U burn-up did not improve the correlation between plutonium production and consumption rates in the joint TR-FR system. Substitution of thorium for natural uranium insignificantly changed ^{233}U consumption rate but remarkably reduced plutonium production rate.

Acknowledgements

The work was carried out within the framework of the State Assignment (project FSWU-2023-0089) with the support of the Ministry of Science and Higher Education of the Russian Federation.

References

- Abagyan LP, Bazazyants NO, Nikolaev MN, Tsybulya AM (1981) Group constants for calculations of nuclear reactors and radiation shielding. Moscow, Energoatomizdat. [In Russian]
- Abagyan LM, Yudkevich MS (1977) Evaluated neutron data for calculating thermal reactors. Issues of atomic science and technology (VANT). Series: Nuclear constants, Issue 2, 21–29.
- Abagyan LP, Prosvetova MP, Yudkevich MS (1989) Neutron data for calculating thermal reactors. VANT, Series: Nuclear Constants, Issue 4, 3–11.
- Alekseev PN, Alekseev SV, Andrianova EF, Asmolov VG, Dekusar VM (2016) Two-component nuclear power system with thermal and fast reactors in closed nuclear fuel cycle. Technosphere, Moscow, 160 pp. [In Russian]
- Alekseev SV, Zaytsev VA (2014) Thorium in Nuclear Power. Technosphere, Moscow, 288 pp. [In Russian]
- Andrianova EA, Davidenko VD, Tsibulsky VF (2008) Computer code DESAE for systematic studies of nuclear power development. Atomic Energy 105(6): 303–307. https://elib.biblioatom.ru/text/atomnaya-energiya_t105-6_2008/p307/ [Accessible from 02.05.2024]
- Andrushechko SA, Afrov AM, Vasil'ev BYu, Generalov VN, Kosourov KB, Semchenkov YuM, Ukraintsev VF (2010) Nuclear power plant with VVER-1000 reactor. From physical fundamentals to evolution of the project. Moscow, Logos, 604 pp. [In Russian]
- Borisov OM, Orlov VV, Naumov VV (2000) Requirements to the reactor core. International Seminar "Fast reactor and fuel cycle with inherent safety for large-scale nuclear power. Fuel balance, economics, safety, wastes, non-proliferation". Moscow, Russia, paper No. 15. [In Russian]
- Dekusar VM, Kagramanyan VS, Kalashnikov AG, Korobeynikov VV, Korobitsyn VE, Klinov DA (2010) Development of mathematical model for nuclear fuel cycle including thermal and fast reactors. Communications of Higher Schools. Nuclear Power Engineering 4: 119–132. https://static.nuclear-power-engineering.ru/journals/2010/04.pdf [Accessible from 02.05.2024]
- Fedorov MS, Zozulya DV, Baidakov NA, Zhiganov AN (2021) Experience of mixed nitride uranium–plutonium fuel fabrication at the Siberian chemical plant jsc site. Nuclear Engineering and Design 384: 1–4. https://doi.org/10.1016/j.nucengdes.2021.111462
- Kagramanyan VS, Kalashnikov AG, Kapranova EN, Puzakov AYu (2014) Comparison of fuel cycles of the nuclear power system based on thermal VVER-TOI reactors and fast BN-1200 reactors. Communications of Higher Schools. Nuclear Power Engineering 4: 92–101. https://doi.org/10.26583/npe.2014.4.10 [Accessible from 02.05.2024]
- Kuzmin AM, Shmelev AN, Apse VA (2015) Modeling of physical processes in nuclear power reactors on fast neutrons. Training manual for higher schools. Moscow, Printing House MPEI. [In Russian]
- [Nuclear Database] JANIS Web (2024) JANIS Web. https://www.oecd-nea.org/janisweb/search/endf [Accessible from 02.05.2024]
- Petrov AYu, Shutikov AV, Ponomarev-Stepnoy NN, Bezzubtsev VS, Bakanov MV, Troyanov VM (2019) Prospects for creation of two-component nuclear power system. Communications of Higher Schools. Nuclear Power Engineering 2: 5–15. https://doi.org/10.26583/npe.2019.2.01 [Accessible from 02.05.2024]
- Ponomarev-Stepnoy NN, Lunin GL, Morozov AG, Kuznetsov VV, Kevrolev VV, Kuznetsov VF (1998) Light-water thorium reactor VVER-T. Atomic Energy 85(4): 263–277. https://elib.biblioatom.ru/text/atomnaya-energiya_t85-4_1998/p263/ [Accessible from 02.05.2024]
- Subbotin SA (2007) Thorium cycle. We choose reactor. Presentation on Science-Technology Council No. 1 of the GC ROSATOM. ProAtom, 11.09.2007. https://www.proatom.ru/modules.php?name=News&file=article&sid=1097 [Accessible from 02.05.2024]
- Zrodnikov AV, Korobeynikov VV, Mosseev AL, Egorov AF, Dekusar VM, Gurdkaya OS, Pupko LP (2021) Analysis of scenarios for full-scale development of two-component nuclear power systems with optimal accounting for export potential of Russian nuclear technologies. Preprint IPPE-3291, Obninsk, Leypunsky Institute of Physics and Power Engineering, 59 pp. https://www.ippe.ru/images/publications/preprints/2021/3291.pdf [Accessible from 02.05.2024]