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ADVANCED CONCEPT OF GAS-COOLED FAST REACTOR-BREEDER OPERATING IN THE NUCLEAR BURNING WAVE MODE

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This paper examines the concept of an innovative gas-cooled fast breeder reactor operating in a self-sustaining mode of slow nuclear burning in the form of traveling wave propagating through a breeding zone. Unique features of this mode, which ensure the reactor's intrinsic safety at its deepest level, are based on the absence of reactivity margin throughout its entire reactor campaign, as well as on a specific mechanism of negative reactivity feedback. The critical state in such a reactor is maintained automatically without any external reactivity control. The fuel for this reactor, except during the startup stage, can be natural or even depleted uranium and thorium. Such a reactor can operate for decades without refueling. Our neutronic calculations were performed on the basis of a deterministic approach using the diffusion approximation of the neutron transport equation in the multiplying medium, together with solving a set of fuel burnup equations. The modified FANTENS-2D computer code was used for the numerical calculations of the steady state of the nuclear burning wave. The fundamental feasibility of implementing a nuclear burning wave mode in a helium-cooled fast reactor is demonstrated. The main parameters of such a reactor were analyzed.

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INTRODUCTION

The sustainable development of global nuclear energy is impossible without addressing its key challenges, such as radically improving reactor safety, ensuring the reproduction of nuclear fuel, strengthening the fissile material non-proliferation regime, and disposing of long-lived nuclear waste. A partial, if not complete, solution to these problems could be provided by implementing the fast breeder reactor concept, which is based on the phenomenon of "slow nuclear burning" or "nuclear deflagration" [1–13].

The idea of long-term operation without refueling of a fast breeder reactor in the mode of self-consistent production and burning of fissile nuclides was formulated back in the late 1950s [1]. Despite repeated attempts, for a long time no one managed to achieve significant progress even at the level of a theoretical solution to this problem. In 1988, Feoktistov [2] proposed an elegant solution to this problem in the form of a traveling "neutron-fission wave", as he called it, propagating in a fertile medium. Feoktistov managed to find an analytical soliton-like solution to the essentially nonlinear equation of neutron diffusion in a multiplying medium, taking into account the self-consistent change in the nuclide composition of the medium under neutron irradiation, which was described by a system of burnup equations of fuel nuclides.

The fundamental possibility of the existence of a traveling neutron-fission wave regime is associated with the gradual spread of the condition for maintaining a nuclear chain reaction, i.e., $k_{inf} > 1$, into the breeding region initially filled with fertile material, for example, ^{238}U, by producing a sufficient amount of fissile nuclide ^{239}Pu in it when irradiated by a neutron flux from the adjacent active zone of the reactor. When the concentration of plutonium in this irradiated zone reaches a certain critical value corresponding to $k_{inf} = 1$, conditions for a chain reaction are created in it, and thus the area of the reactor active zone expands into the breeding zone.

The characteristic size of this region is determined by the free path of a fast neutron in such a medium and is on the order of several centimeters. The characteristic time to reach critical conditions is determined mainly by the half-life of the beta decay of ^{239}Np, that is, the second intermediate nuclide in the corresponding nuclear transformation chain (Fig. 1), and it is about two and a half days. With the accumulation of fission products and the corresponding increase in the total macroscopic cross-section of neutron absorption by all nuclides of the core, the chain reaction in the "old" part of the core will be suppressed, and thus a gradual shift of the active zone to the breeding zone will be observed.

From a physical point of view, the shift of the active zone described above is a complete analogy to the process of slow chemical burning, or the so-called deflagration phenomenon. This phenomenon occurs, for example, when igniting a raw log from one end, when the rapid spread of the flame along the log is impossible due to its high moisture content. However, due to thermal conductivity, moisture evaporates from the layer closest to the burning front, and thus the flame gradually moves along the log. If the medium is prepared for burning in advance, then the propagation of the burning front occurs in the form of a shock wave. This phenomenon is called detonation. An analogue of chemical detonation in the case of "nuclear burning" is a nuclear explosion, that is, an uncontrolled chain reaction in a supercritical system, when the concentration of fissile nuclides is higher than the critical value.

In contrast to the process of chemical burning, in the case of nuclear burning, "medium preparation" occurs through neutron irradiation of the nearby layer of fertile material and the production of fissile nuclides in it until their critical concentration is reached. The phenomenon of "nuclear deflagration" with a very slow production of fissile nuclides (about 2.36 days), and almost

instantaneous burning of their excess on the characteristic time scales of the development of a chain reaction in microseconds, can potentially ensure the automatic maintenance of the critical state in such a breeder reactor for a long time and without any reactivity margin. However, the physics of this phenomenon requires detailed study.

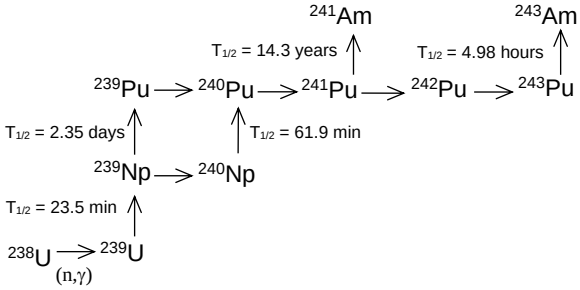


Fig. 1. Nuclear transformations chain for U-Pu cycle

In order to find an analytical solution to such a complex system of nonlinear equations, Feoktistov had to make a number of significant simplifications to the mathematical model. Among the simplifications used: 1) one-dimensional geometry, that is, the propagation of a nuclear burning wave in an infinite half-space, which means the ignoring transverse neutron leakage; 2) a one-group approximation that ignores the dependence of the cross-sections of nuclear processes on the energy of neutrons; 3) maximum simplification of the chain of nuclear transformations of fuel nuclides, limited to only three of them ^{238}U, ^{239}Np and ^{239}Pu; 4) replacement of fission products with ^{238}U nuclei, when calculating macroscopic neutron absorption cross-sections, among others. In addition, Feoktistov looked for a self-similar solution describing the one-dimensional spatial distributions of the neutron flux and the above-mentioned nuclides in the reference system accompanying the traveling wave, thereby solving a stationary problem. This allowed him to find an analytical solution to this system of equations and establish a physical criterion for the existence of a traveling wave regime of slow nuclear burning in such a system, as well as to show that such a regime is possible, but only in a fast reactor with a sufficiently hard neutron spectrum and impossible in a thermal one [2].

In subsequent studies, other authors proposed various approaches and computational methods for describing the phenomenon of nuclear deflagration, using various names for such a reactor: Traveling Wave Reactor [3, 4], the CANDLE [5], and the Nuclear Burning Wave (NBW) [6–13]. These methods made it possible to take into account the influence of such factors as transverse neutron leakage, a more complete chain of nuclear transformations in such a system, the possibility of implementing the NBW mode in the uranium-thorium fuel cycle, among others.

One of the most important aspects of studying the NBW phenomenon, namely the study of its stability [10] and various types of transient processes in such a reactor [11], requires solving a truly non-stationary problem. It is precisely this approach that was consistently developed in our works [6–13], step by step improving the computational model and bringing it closer to the realistic core of the NBW reactor.

Our research results have shown that the NBW mode is remarkably stable, thanks to the presence of a unique negative reactivity feedback mechanism specific to this mode [7]. Another important result was the determination of the minimum (critical) radius of a cylindrical NBW reactor, the value of which is determined by the maximum permissible transverse neutron leakage to maintain the NBW mode. Using uranium-plutonium fuel, the critical radius of the NBW reactor, according to our calculations, is approximately 115 cm [8]. The use of a radial reflector allows this value to be reduced somewhat [13].

The NBW propagation velocity increases rapidly with increasing core radius, as does the neutron flux and reactor power [8]. When choosing a minimum cylindrical core radius, and therefore a minimum reactor power, the wave propagation velocity is several centimeters per year. The minimum thermal power of such a reactor, according to preliminary calculations, is approximately 1 GW. As follows from our calculations, the characteristic longitudinal size of the nuclear burn region in such a reactor with uranium-plutonium fuel is just over two meters. This means that in order to ensure long-term (decades) maintenance of the stationary mode of NBW propagation in such a reactor, the total length of its core (ignition zone and breeding zone, Fig. 2) must be several meters. The technical difficulties with pumping a liquid metal coolant such as lead-bismuth eutectics or sodium, which can arise in cooling systems of this length, stimulated us to consider the option of an NBW reactor with a gas coolant. The main difference from our previous neutronics calculations for reactors with Pb-Bi eutectics or Na as a coolant will be, in addition to changes in the corresponding nuclear constants, changes in the ratio of volume fractions of fuel, coolant and structural materials for reactors of various types. This work is devoted to the study of the possibility of implementing the NBW mode in a helium-cooled reactor, as well as determining the main parameters of such a reactor.

1. CALCULATION SCHEME

The original deterministic computer code FANTENS-2D was developed in our works [6–13] specifically for a detailed study of the phenomenon of slow nuclear burning in a multiplying medium. It allows us to simulate the initiation and evolution of a nuclear burning wave [7, 9, 12], as well as to study the stability of the NBW mode [10], various transient processes in such a reactor, such as forced shutdown of the reactor with subsequent restart [11], control of its power [13, 14], and reactor behavior in various types of severe accidents. The FANTENS-2D code is based on a numerical solution of a multigroup non-stationary neutron diffusion equation in a cylindrical reactor consisting of several initially homogeneous zones (see Fig. 2), including an ignition zone of radius R and length L_{ign} containing enriched nuclear fuel for initiating the NBW mode, a breeding zone of the same radius R and length (L - L_{ign}) filled with ^{238}U or ^{232}Th fertile material (or a mixture thereof), and a radial reflector zone along the whole reactor of a thickness W_{ref}. In addition, the calculations

take into account the presence of coolant and structural materials in all these zones in realistic proportions of the corresponding volume fractions, depending on the type of fuel used, the type of coolant and the reflector material.

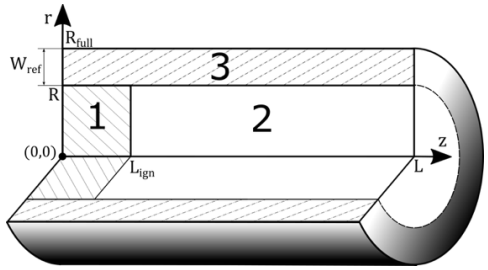


Fig. 2. Scheme of a cylindrical reactor with ignition zone 1; breeding zone 2, and radial reflector 3

To simulate the phenomenon of slow nuclear burning, simultaneously with solving the neutron diffusion equation, it is necessary to solve the system of fuel burnup equations, according to the nuclear transformations chain of fuel nuclides shown in Fig. 1. This makes this problem essentially nonlinear, but the continuous change in the nuclide composition of the reactor core is the determining factor in the formation of the NBW mode. A significant challenge in modeling transient processes in a NBW reactor is the need to simultaneously account for interrelated nuclear processes occurring on significantly different time scales, from fractions of a microsecond during the development of a nuclear chain reaction to the characteristic times of delayed neutron appearance (tens of seconds) and fuel nuclide transmutations (days and months). All of this combined leads to significant computer time consumption during these calculations. To solve this problem, the FANTENS-2D code uses a calculation scheme with a variable time step, the choice of which is determined by the desire to speed up calculations while ensuring the required accuracy of numerical calculations. A detailed description of the numerical calculations scheme used in the FANTENS 2D code is given in [13].

To answer the question of the existence of the NBW mode in a helium-cooled fast reactor, it is not necessary to solve the non-stationary problem for describing transient processes [11] using the entire arsenal of capabilities of the FANTENS-2D code. Therefore, at the first stage of our study, following Feoktistov [2], we will limit ourselves to the search for a self-similar solution to the neutron diffusion equation. However, unlike in the work [2], we will be looking for a solution in a two-dimensional cylindrical geometry (see Fig. 2), with a more complete chain of nuclear transformation for fuel nuclides, presented in Fig. 1, including nuclear fission fragments and using a multigroup approach to adequately account for the dependence of all cross-sections on neutron energy. For this purpose, we will use some parts of the FANTENS-2D code, adapting them to find a self-similar solution to the two-dimensional multigroup diffusion equations of neutron transport for the cylindrical geometry [8, 13]:

$$\frac{1}{\upsilon^g}\frac{\partial\Phi^g}{\partial t}+\frac{1}{r}\frac{\partial}{\partial r}rV_r^g+\frac{\partial}{\partial z}V_z^g+$$
$$+\left(\Sigma_a^g+\Sigma_{in}^g+\Sigma_{\mathrm{mod}}^g+\Sigma_{in}^g\right)\Phi^g-$$
$$-\Sigma_{\mathrm{mod}}^{g-1}\Phi^{g-1}-\sum_{g'=1}^{g-1}\Sigma_{in}^{g'\to g}\Phi^{g'}-$$
$$-\frac{1}{k_{eff}}\left[\begin{array}{l}\chi_f^g\sum_{g'=1}^{G}\left(\nu_f\Sigma_f^g\right)^{g'}\Phi^{g'}+\\+\sum_j\chi_d^{gj}\sum_l\beta_l^j\sum_{g'=1}^{G}\left(\nu_f\Sigma_f^g\right)^{g'}\Phi^{g'}-\\-\sum_j\chi_d^{gj}\sum_l\lambda_l^jC_l^j\end{array}\right] \quad (1)$$
$$=Q^g(r,z,t);$$
$$V_r^g=-D^g\frac{\partial\Phi}{\partial r};\ V_z^g=-D^g\frac{\partial\Phi}{\partial z}.$$

Here $\Phi^g(r,z,t)$, $V_r^g(r,z,t)$ and $V_z^g(r,z,t)$ are the scalar flux and the radial and axial components of the vector neutron flux in the energy group g; $D^g=1/3\Sigma_{tr}^g$ is the group diffusion coefficient; $\Sigma_a^g=\Sigma_c^g+\Sigma_f^g$, Σ_f^g, Σ_c^g, Σ_{in}^g, Σ_{mod}^g, and Σ_{tr}^g, are the macroscopic cross-sections for: absorption, fission, radiative capture, inelastic neutron scattering, moderation, and transport, respectively; $\Sigma_{in}^{g'\to g}$ is the macroscopic cross-section for the inelastic neutron scattering with transition from group g' to g; χ_f^g is the neutron fission group spectrum; ν_f^g is the average number neutrons in the group *g* per fission; υ_f^g is the neutron group velocity; β_l^j, C_l^j, and λ_l^j are the fraction of delayed neutrons, the concentration, and decay constant of precursor nuclei in the *j*-th group for the *l*-th fissile nuclide; χ_d^{gj} is the delayed-neutron energy spectrum for the *j*-th group of precursors (it is assumed that the delayed-neutron energy spectra are identical for all fissile nuclides); $Q^g(r,z,t)$ is the above-mentioned volume neutron source in the group *g*; k_{eff} is the effective neutron multiplication factor.

The change of concentrations of the nuclides in the transformation chain shown in Fig. 1 can be described by the following equations [13]:

$$\frac{\partial N_1}{\partial t}=-\sum_g\sigma_{a1}^g\Phi^gN_1;\quad \frac{\partial N_9}{\partial t}=\Lambda_6N_6;$$
$$\frac{\partial N_{10}}{\partial t}=\sum_{l=1,4,5,6,7}\sum_g\sigma_{fl}^g\Phi^gN_l; \quad (2)$$
$$\frac{\partial N_l}{\partial t}=\left(\sum_g\sigma_{al}^g\Phi^g+\Lambda_l\right)N_l+\left(\sum_g\sigma_{c(l-1)}^g\Phi^g+\Lambda_{l-1}\right)N_{l-1}$$
$$+\delta_{l5}\sum_g\sigma_{c3}^g\Phi^gN_3,\quad (l=2...8),$$

where $\sigma_{al}^g=\sigma_{cl}^g+\sigma_{fl}^g$, σ_{cl}^g, σ_{fl}^g are the group microscopic cross sections for absorption, radiative neutron capture, and fission of the *l*-th nuclide in the group *g*, respectively; $\Lambda_l=\ln 2/T_{1/2}^l$ is the β-decay

constant of the *l*-th nucleus, and $T_{1/2}^{l}$ is the corresponding half-life; δ_{l5} in (2) is the Kronecker symbol. The constants Λ_2, Λ_3, and Λ_6 are considered to be non-zero, while $\Lambda_4 = \Lambda_5 = \Lambda_7 = \Lambda_8 = 0$. In this scheme, we neglect the burnup of ^{239}U, ^{241}Am, and ^{243}Am nuclei ($\sigma_{a2}^{g} = \sigma_{a8}^{g} = \sigma_{a9}^{g} = 0$), because the decrease of their concentrations due to the absorption reactions is small, as compared to the considered processes. The change in the neutron fission products due to the neutron absorption is neglected here.

For simplification, in considered calculation scheme is adopted that the nuclides ^{239}Np and ^{242}Pu, after a neutron capture, are immediately transformed into ^{240}Pu and ^{243}Am and we drop the intermediate nuclei ^{240}Np and ^{243}Pu from consideration. As a result, we include in the main chain of transformations 10 nuclei, whose numbering used in the fuel burnup equations is given in Table. Five nuclei with numbers 1, 3, 4, 5, 6, 7 are considered as fissile ones. The pair of fragments produced in each fission event is considered in our calculations as one nucleus, denoted as FP [14].

Nuclide numbering of the U-Pu cycle

No.	1	2	3	4	5
Nucleus	^{238}U	^{239}U	^{239}Np	^{239}Pu	^{240}Pu
No.	6	7	8	9	10
Nucleus	^{241}Pu	^{242}Pu	^{243}Am	^{241}Am	FP

The equations of burnup and production of the fuel nuclides for the U-Pu cycle can be presented in a general form as:

$$\frac{\partial N_i}{\partial t} = -\Lambda_i N_i + \sum_j \Lambda_i^j N_j \,, \qquad (3)$$

where $\Lambda_i = \sigma_{ai}\Phi + \lambda_i$ is the transmutation reaction rate of nuclide *i* and Λ_i^j is the transmutation reaction rate of nuclide *j* to nuclide *i*.

When moving to self-similar coordinates:

$$x = z + Vt \;; \qquad (4)$$

$$\frac{\partial}{\partial t} = V\frac{\partial}{\partial z}; \quad \frac{\partial}{\partial z} = \frac{\partial}{\partial z} \,, \qquad (5)$$

where *V* is the velocity of the NBW motion along the axis z, the equation (3) takes the form:

$$V\frac{\partial N_i}{\partial x} = -\Lambda_i N_i + \sum_j \Lambda_i^j N_j \,. \qquad (6)$$

Such a system cannot be solved analytically in the general case, so we will look for solutions using numerical methods.

The solution of equation (1) gives us the form of the neutron flux distribution $\Phi(r,z) \equiv \Phi$ and k_{eff} as the eigenvalue of this equation for the given distributions of nuclides in the volume of the active zone, but does not give the absolute value of the flux distribution amplitude $A_\Phi (\Phi = A_\Phi \phi)$.

The solution of equation (6) yields the distribution $N_i(r,x)$ of main fuel nuclides and fission products in the NBW region propagating at velocity V into the breeding zone initially filled with fertile material ^{238}U.

In steady-state conditions, the nuclear burn wave propagates with constant velocity without changing the shape and amplitude of the neutron flux distribution. The NBW reactor power is an internal parameter of the system and is completely determined by its composition and geometry of an active zone define by the NBW shape. Consequently, the self-similar solution of equations (1) and (6) should yield neutron flux and velocity values such that the nuclide distribution from the solution of equation (6), substituted into equation (1), yields an effective neutron multiplication factor k_{eff} equal to unity. This is the main idea of the iteration procedure we used for finding the main parameters of the steady-state mode of the NBW reactor with helium coolant that we are considering.

Below is a scheme of the iterative procedure we use to find a self-similar solution to the problem under consideration:

1). For a given initial nuclide distribution $N_i^{(0)}(r,x)$, we solve equation (1), from which we obtain k_{eff} and the neutron flux shape $\phi^{(0)}(r,x)$, $A_\Phi^{(0)} = 0$ or the value from the previous iterations, see step 3. The position of the neutron flux maximum is placed at the center of the spatial region under consideration.

2). For the obtained neutron flux distribution $A_\Phi^{(0)} \cdot \phi^{(0)}(r,x)$ and a chosen value of the NBW velocity V, we solve equation (6), from which we obtain a new nuclide distribution $N_i^{(1)}(r,x)$.

3). For the new nuclide distribution $N_i^{(1)}(r,x)$, we repeat step 1. If k_{eff} differs from 1, we change the neutron flux distribution amplitude obtained from step 1 to $A_\Phi^{(1)} = \alpha A_\Phi^{(0)}$ (α is selected based on the obtained value of k_{eff}), and repeat step 2.

4). We repeat step 3 until k_{eff} approaches 1 with the required accuracy. When calculating k_{eff}, $\phi^{(n)}(r,x)$ will also be obtained. Once this condition is met, the resulting nuclide distribution $N_i^{(n)}(r,x)$ will become the new initial distribution for step 1.

5). The neutron flux distributions at step 1 $\phi_{sh}^{(0)}(r,x)$ and at step 4 $\phi^{(n)}(r,x)$ can be differ in shape and the position of the maximum. We'll repeat steps 1–4 until these differences become small enough.

6). We'll change the velocity V and repeat steps 1–5.

7). We'll repeat steps 1–6 until the difference in the flow distributions from step 5 becomes negligible.

The block-scheme of this iterative procedure looks is represented in Fig. 3.

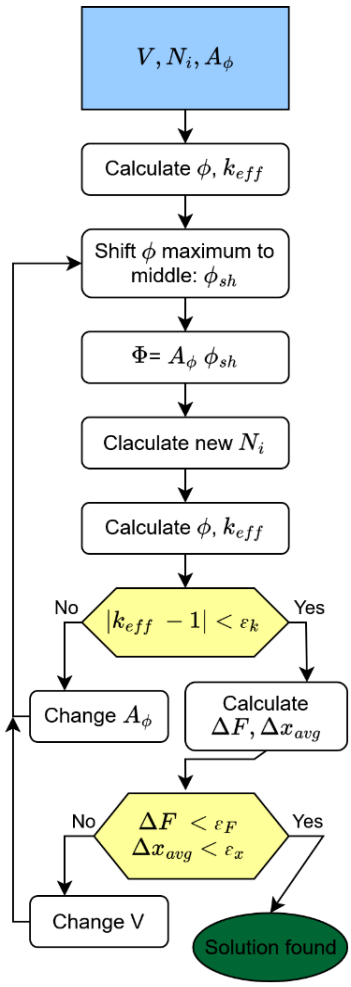


Fig. 3. Diagram of the iterative procedure used to find a self-similar solution for the steady-state NBW mode

RESULTS AND DISCUSSION

When selecting the starting configuration of the reactor core (composition and dimensions) for the iteration procedure, the parameters of the ALLEGRO helium-cooled fast reactor currently under development were used [15]. However, instead of the uranium-plutonium carbide used in ALLEGRO, uranium-zirconium metal fuel UZr (10% wt.) was chosen as fuel, since it was not possible to obtain a self-similar solution for the NBW reactor in the case of uranium-plutonium carbide fuel. This is apparently due to a decrease in the efficiency of the breeding process due to the softening of the neutron spectrum during scattering on relatively light carbon nuclei. For the same reason, Zircaloy 4 was chosen instead of silicon carbide SiC as the structural material of the reactor core (fuel cladding). As in ALLEGRO, helium gas under a pressure of 70 atmospheres was assumed as the coolant in our calculations, and silicon carbide SiC (0.7:0.3) was used as the reflector. The volume fraction ratio of fuel is 42% (with 20% porosity), coolant is 38.3%, and the remaining volume of 19.2% is for structural materials, approximately corresponding to the ratio in the ALLEGRO reactor. It was assumed that the radial reflector (see Fig. 2) consists of 78.3% silicon carbide and is cooled by helium under a pressure of 10 atmospheres and a relative volume fraction of 21.3%. The geometric dimensions of the reactor in the calculations were taken as follows (see Fig. 2): reactor length L = 5 m, core radius R = 126 cm, radial reflector thickness W_{ref} = 71 cm. Recall that each of the reactor zones is considered initially homogeneous.

Fig. 4 shows the results of the numerical calculation of the axial distributions of the neutron flux Φ and the concentrations of the fertile material ^{238}U, the main fuel nuclide ^{239}Pu, and the fission products FP along the axis of the cylindrical reactor near the center of the wave (the maximum of neutron flux), obtained using the above-described scheme for calculating the steady state of the NBW propagating from right to left. The wave velocity for this case is $V \approx 10$ cm/year, thermal power of such a reactor is $P \approx 3.6$ GW.

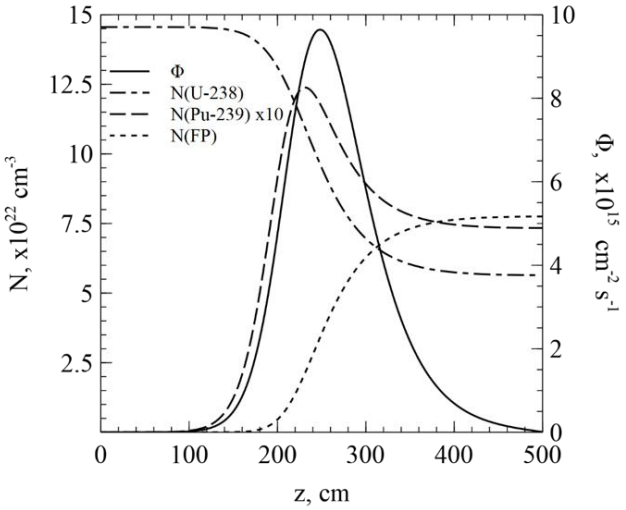


Fig. 4. The axial distributions of neutron flux Φ, concentrations of fertile material ^{238}U, main fuel nuclides ^{239}Pu and fission products FP in the NBW propagating from right to left

If we define the nuclear burning "wavelength" in such a reactor as the size of the region along the direction of wave propagation (axis z) within which the neutron flux is at least 0.1 times its maximum value, then, as follows from Fig. 4, this size is approximately 2 m. The curve describing the distribution of the main fissile nuclide ^{239}Pu is approximately 10 cm ahead of the neutron flux curve at the wave motion, as expected, according to the physics of the nuclear deflagration phenomenon discussed in the Introduction. The ^{239}Pu concentration near the neutron flux maximum reaches approximately 8% of all fuel nuclides, including ^{238}U and all produced MA. Behind the wave, the ^{239}Pu concentration drops to 5%, reaching a plateau. This means that after the NBW has passed, the content of ^{239}Pu in the entire reactor volume ("spent fuel") is about 5%. Taking into account the entire plutonium isotope vector, as well as all other MA (see Table), the content of fissile nuclides in the spent fuel of the NBW reactor will be approximately 8%, and approximately 10% at the peak of the wave. The burnup of ^{238}U, which is the actual nuclear fuel for the NBW reactor, is approximately 60% (see Fig. 4).

The two-dimensional (radial and axial) distribution of the neutron flux Φ(r, z) and concentrations of ^{239}Pu, ^{238}U and fission products FP are shown in Fig. 5.

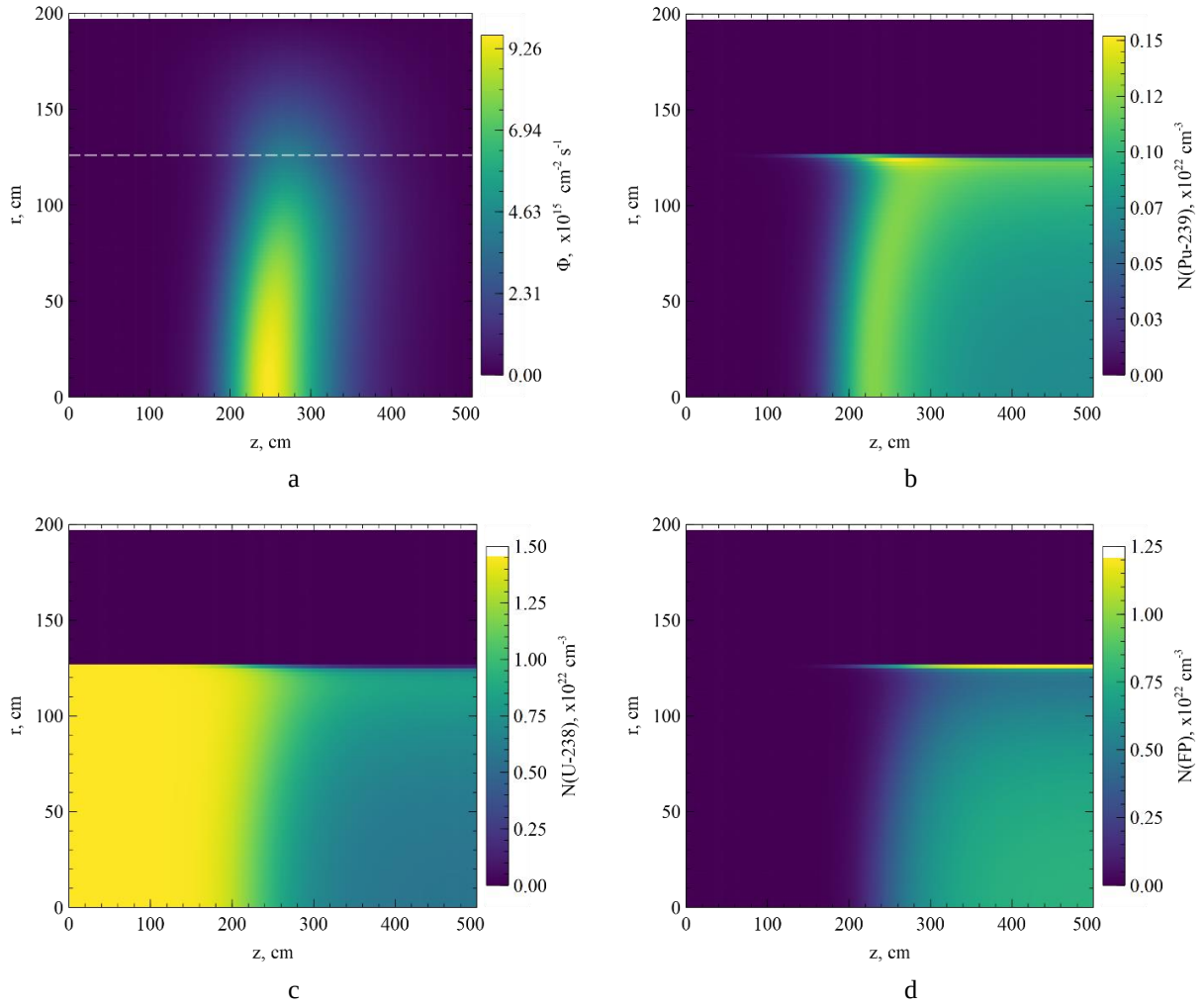


Fig. 5. Radial (along r) and axial (along z) distribution of the neutron flux Φ (a) and concentrations of ^{239}Pu (b), ^{238}U (c), and fission products FP (d) when the NBW passes from right to left along the axial direction (z axis) of the cylindrical reactor core of radius R = 126 cm with the radial reflector of thickness W_{ref} = 71 cm. The dashed line in Fig. 5,a shows the position of the radial reflector

A combined analysis of the neutron flux distribution in the NBW presented in Fig. 5, along with the corresponding distributions of the main nuclides ^{239}Pu, ^{238}U and fission products FP forming and burning in the reactor core under the neutron irradiation, reveals nontrivial features in the region adjacent to the radial reflector. These features are associated with the use of relatively lightweight silicon carbide SiC as a reflector in a fast-spectrum reactor.

Despite the fact that, as can be seen from Fig. 5,a, the neutron flux near the reflector is significantly weaker than in the central part of the reactor core, we observe an elevated Pu concentration near the reflector, which then drops sharply at the boundary of the core and reflector (see Fig. 5,b). The presence of such a "parapet" in the Pu distribution near the boundary of the reactor core was also observed in our calculations with an "open boundary", i.e., without a reflector [7]. This phenomenon is obviously associated with neutron leakage through the lateral boundary of the cylindrical core. In a self-consistent process of fissile nuclides production and burnup, such as a nuclear burning wave, the Feoktistov criterion [2] must be satisfied. According to this criterion, the NBW regime is possible if the equilibrium concentration of fissile nuclides in the nuclear burning region exceeds their critical concentration, which ensures the maintenance of a nuclear chain reaction. The localized reduction in neutron flux due to leakage at the core boundary is automatically compensated for by an increase in the concentration of fissile nuclides in this region.

At the very edge of the core at the boundary with the reflector, increased ^{238}U burnup (see Fig. 5,c) and FP production (see Fig. 5,d) are observed. The reason is the increase in the fission cross-section of Pu with the softening of the neutron spectrum in this region due to the return of neutrons to the active zone after moderation in the SiC reflector. This effect was not observed when using a reflector made of heavy materials, such as lead-bismuth eutectic [13].

CONCLUSIONS

Based on the solution of the self-similar problem of neutron transport in a multiplying medium using the FANTENS-2D computer code [13], the possibility of the existence of a self-sustaining regime of a nuclear burning wave (NBW) in a fast reactor with helium coolant is shown. For computer modeling of the NBW mode in

such a reactor, we used parameters similar to those of the core of the ALLEGRO gas-cooled fast reactor currently under development [15].

To improve the efficiency of the breeding process, which is critical for the NBW reactor, it is proposed to use metallic uranium-zirconium fuel instead of uranium carbide in ALLEGRO. For the same reason, it is proposed to use Zircaloy-4 as a structural material instead of silicon carbide in ALLEGRO. As in the ALLEGRO reactor, silicon carbide is proposed as the radial reflector material for the NBW reactor. This leads to some alignment of the energy production front at NBW propagation by softening the neutron spectrum and correspondingly increasing the burnup cross-section for fissile nuclides near the radial reflector, partially compensating the decline in neutron flux at the periphery of the core.

Future studies plan to optimize the parameters of the proposed helium-cooled NBW reactor with the goal of reducing its power and neutron fluence to the lowest possible levels. We also plan to study the possibility of creating a gas-cooled NBW reactor with mixed uranium-thorium fuel.

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ПЕРСПЕКТИВНА КОНЦЕПЦІЯ ГАЗООХОЛОДЖУВАНОГО ШВИДКОГО РЕАКТОРА-БРИДЕРА, ЩО ПРАЦЮЄ В РЕЖИМІ ХВИЛІ ЯДЕРНОГО ГОРІННЯ

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Розглядається концепція інноваційного реактора-розмножувача на швидких нейтронах із газовим охолодженням, що працює в самопідтримному режимі повільного ядерного горіння у формі хвилі, що біжить та поширюється крізь зону розмноження. Унікальні особливості цього режиму, що забезпечують власну безпеку реактора на найглибшому рівні, базуються на відсутності запасу реактивності протягом усієї реакторної кампанії, а також на специфічному механізмі негативного зворотного зв’язку за реактивністю. Критичний стан у такому реакторі підтримується автоматично без будь-якого зовнішнього контролю реактивності. Паливом для цього реактора, крім стадії запуску, може бути природний або навіть збіднений уран і торій. Такий реактор може працювати десятиріччями без перезавантаження. Наші нейтронні розрахунки були виконані на основі детерміністичного підходу з використанням дифузійного наближення рівняння переносу нейтронів у мультиплікуючому середовищі разом із розв'язанням системи рівнянь вигоряння палива. Для числових розрахунків стаціонарного стану хвилі ядерного горіння було використано модифікований комп'ютерний код FANTENS-2D. Продемонстровано принципову можливість реалізації режиму хвилі ядерного горіння у швидкому реакторі з гелієвим охолодженням. Проаналізовано основні можливі параметри такого реактора.