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Particularities of spatial kinetics of hybrid thorium reactor installation containing the long neutron source based on magnetic trap

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Abstract. In this work, we study the features of the spatial kinetics of installation as a hybrid thorium reactor with an elongated plasma neutron source based on a magnetic trap. The active zone of the installation under study consists of an assembly of hexagonal fuel blocks of a unified design and a long solenoid with a high-temperature plasma column passing through the axial region of the core. Combining engineering expertise in creating nuclear reactors with a physics-technical potential for obtaining high-temperature plasma in a long magnetic trap we ensure the solution of the multidisciplinary problem posed. These studies are of undoubted practical interest, since they are necessary to substantiate the safety of operation of such hybrid systems. The research results will allow optimizing the active zone of the hybrid system with leveling the resulting offset radial and axial energy release distributions. Results of our study will be the basis for the development of new and improvement of existing methods of criticality control in related systems such as “pulsed neutron source – subcritical fuel assembly”

1. Current state of the art

The study of physics of low-power high temperature gas-cooled thorium reactor unit (HGTRU) (Tomsk Polytechnic University (TPU), Tomsk) was started three years ago [1–3] and its authors have already chosen the most appropriate core configuration and a nuclear fuel material composition. The authors have stated [1] that the reactor being studied is capable of operating for not less than 3000 effective days at the power of 60 MW_{th}.

A special feature of the reactor studied in the work [1, 3] is that it is able to produce heat, electricity energy and hydrogen simultaneously, and its core can rather easily be modified for solving another task [4]. Besides, the reactor power can be adjusted according to the regional energy needs.

At present a series of experiments for the purpose of studying physical properties of the new generation nuclear fuel for HGTRU is being carried out at TPU. Dispersion fuel, which is being developed at TPU, also refers to the fuel of new generation. It is a fuel material with advanced mechanical, thermophysical and neutronic properties. As far as the reactor core with such fuel has not been studied earlier in neutronic experiments, on this reason it became necessary to create an installation to carry out required experiments. A special facility intended for studying neutronic properties of the dispersion (Th,Pu)O_2 fuel was suggested in the capacity of such installation by the employee group of Budker Institute of Nuclear Physics of SB RAS (Novosibirsk, Russia) [4–6]. This facility is an assembly of fuel blocks the axial part of which is substituted by a long magnetic trap [5,6] with high temperature plasma providing generation of thermonuclear neutrons.

In our paper, we present the results of computer simulation of neutronic processes in the core of the HGTRU for 30 different loading options. To provide stable reactor operation the fraction of the dispersion phase (ω_f) and the composition of the starting fissionable nuclide were chosen. The possibility to modify the axial part of the core was investigated in accordance with the concept suggested in the works [4].

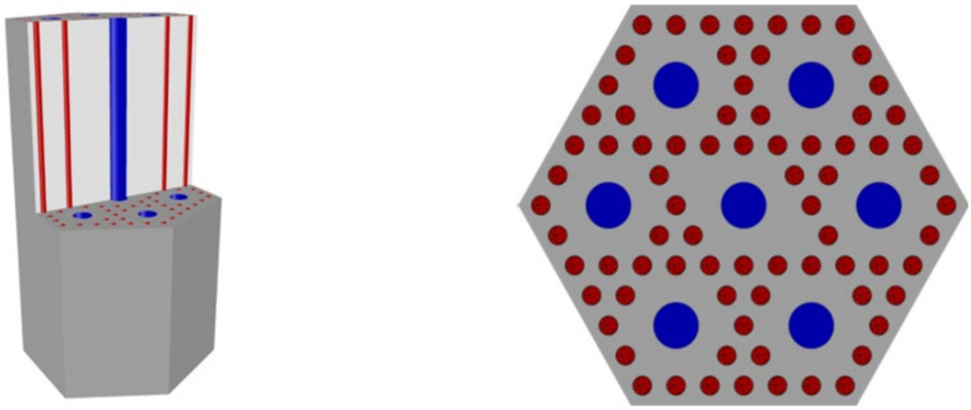
From the perspective of advancement in the field of fundamental knowledge, the purpose of these researches is to expand and to deepen some understanding of the possibilities that become open due to the development of thermonuclear power energy technologies of the future. From the point of view of important applied tasks solution it can be said that the results will form the basis of organizing a stable operation of HGTRU in a long-term operating cycle (not less than 10 years) with a high burn-up degree of both thorium and plutonium components of the fuel.

2. Numerical investigation

2.1. Computational model of the core basic configuration

The core of the HTGRU [1, 3] consists of fuel blocks having the configuration of a graphite hexagon. The core is surrounded by two rows of graphite hexagons of the same configuration, but without holes for fuel. On the top and on the bottom the core is closed with graphite blocks too, but they are laid in one row there.

Schematic drawing of the fuel block and its cross section are presented in Figure 1*a, b*. One can see that there are the cylindrical channels along all length of the block. The fuel block contains 76 channels of small diameter for fuel pellets (red channels of small diameter) and seven channels of large diameter for helium flows (blue channels of large diameter).



(a) (b)

Figure 1. The schematic of the core fuel block (a) and its cross section (b).

Microencapsulated fuel (microfuel) for fuel pellets of HGTRU [1–3] is a spherical fuel kernel from fissionable material (Th,Pu)O_2 with the diameter of 350 μm, covered successively by the layers of PyC and Ti_3SiC_2. These fuel kernels are dispersed into a graphite matrix of cylindrical fuel pellets, which are placed in the fuel block.

At the first stage of research presented in the works [1,3], three types of fuel pellets with reference designations 0817, 1017 and 1200 were used. The diameters of these fuel pellet types were different:

8.17×10^{-3}, 10.17×10^{-3} and 12.00×10^{-3} m, correspondingly. At the same time, the height of the fuel pellets (20×10^{-3} m) and thickness of the external SiC layer (0.3×10^{-3} m) were equal for all types.

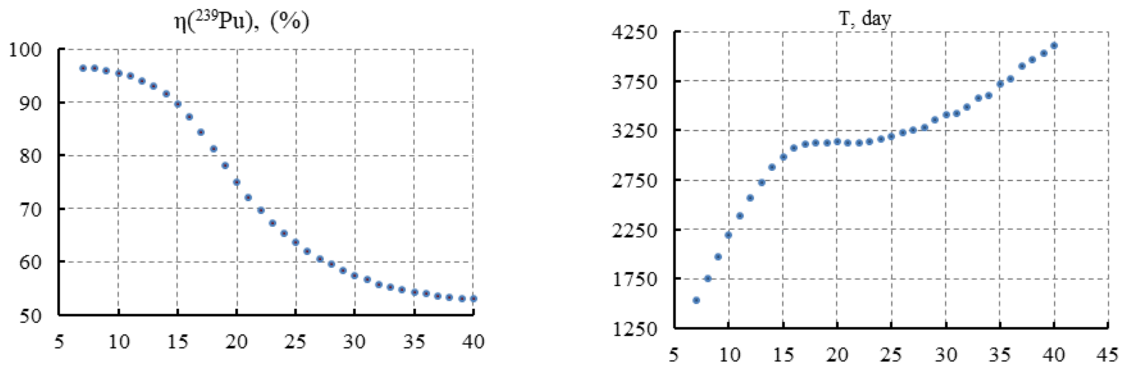
In the work [1] it was shown that from the point of view of the campaign life the best reactor core loading can be achieved by using the fuel pellets of the type 1200. However, in the reactor with the fuel pellets of this type the increase of the disperse phase quantity ω_f ($\omega_f = V_{coated\ particle} \times N / V_{matrix}$) results in significant decrease of spent $((\eta(^{239}Pu) = (N(t_{start}) - N(t_{end}))/N(t_{start}))$ ^{239}Pu. Therefore, the fuel pellet of the type 1017 was chosen in the researches for better ^{239}Pu burnup.

2.2. Neutronic computations for different core loading configurations

At the second stage of the performed computer calculations the cylindrical reactor cell that is an equivalent system of Wigner-Seitz, is studied. The fuel part of the cell is homogenized; computations are performed in the program WIMS-D5B [7].

WIMS-D5B (Nuclear Energy Agency) is a universal program for computing cells of different reactor types. The WIMS-D code uses a 69-group system of constants on the basis of the evaluated nuclear data base ENDF/B-VII.0 (The basic library has been compiled with 14 fast groups, 13 resonance groups and 42 thermal groups), which allows computing reactors on fast and thermal neutrons. In WIMS-D, the accuracy of the computed functionals is set by the step equal to 10 days. To assess poisoning the reactor by xenon, the first 2 days are computed separately.

To determine effective K-factor (k_{eff}) an axial and radial geometrical parameter (B) is introduced. It is computed taking into account transition from the actual core size to the equivalent system of Wigner-Seitz. "White mirror" is used as boundary conditions on the side surface of the cell and "translation symmetry" is used on the cell faces. The computation results of 30 core-loading options with fuel pellets of the type 1017 are presented in Figure 2.



(a) (b)

Figure 2. Neutronic properties of the equivalent system: *(a)* – dependence of $\eta(^{239}Pu)$ on ω_f, *(b)* – dependence of irradiation time T on ω_f

The results of computer simulation show that the increase of ω_f by more than 17−18 % does not lead to any noticeable increase of the reactor fuel campaign T (days) (see Figure 2*b*). The increase in ω_f is also pointless for the spectrum $\varphi_{Vn}(E)$ of the reactor under study [1].

Therefore for further research we chose the fuel pellet of the type 1017 with $\omega_f = 17$ %. According to the results obtained in WIMS-D, the reactor with such fuel pellet is capable of operating for 3110 effective days at the power P = 60 MW_{th}. The neutron flux density in the core φ_{Vn} is 8.14×10^{13} $n\cdot cm^{-2}\cdot s^{-1}$. There are 9.41% of neutrons with the energy E_n up to 4eV, 35.57 % of neutrons with E_n from 4 eV up to 183.2 keV, and 55.02 % of neutrons with E_n from 183.2 keV to 10.5 MeV. In this conditions, k_{eff} is 1.24, $\rho_{initial}$ is $(k_{eff} - 1)/k_{eff} \approx 19.35$ %, $(d\rho/dt) \approx 6.2 \times 10^{-3}$%/days. The burnup of ^{239}Pu is 85 % ($\eta = N(0) - N(3110))/N(0)$), and the burnup of ^{232}Th is 9.66 %. The computation results of fuel isotopic composition evolution $N(t)$ are presented in Figure 3.

Additionally, concentrations change dependence of isotopes ^{231,233}Pa on time $^{231,233}N(t)$ were derived and concentrations of ^{149}Sm и ^{135}Xe were analyzed. The data of ^{231,233}Pa, ^{233}U, ^{149}Sm и ^{135}Xe concentrations will allow evaluating the required intensity $I_n(t)$ of neutron emission from a thermonuclear reaction in a plasma column. By varying percentage of DT-reactions in comparison with DD-reactions in the plasma column, one can change I_n within the range from 10^{13} n·s^{-1}·cm^{-1} to 2×10^{14} n·s^{-1}·cm^{-1}. That should provide the condition k_{eff} = constant during both the first operation days of the facility and the whole fuel irradiation cycle.

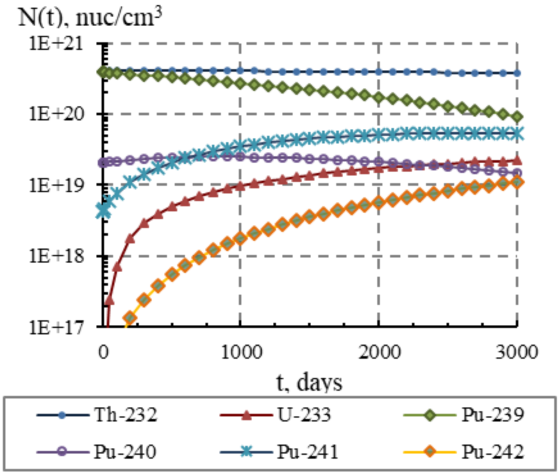


Figure 3. Fuel isotopic composition evolution in HGTRU with the fuel pellet of the type 1017.

The burnable absorber can compensate large initial reactivity excess $\rho_{initial}$ ($\rho_{initial}$=19.35%). For compensation of $\rho_{initial}$ in the LWR and BN-type reactors, for example, Gd_2O_3, Er_2O_3 and B_4C are used. In this case daughter nuclides formed as a result of radiation capture reaction on ^{157}Gd, ^{167}Er and ^{10}B do not have significant impact from the point of view of further neutronic processes in the core.

In solving the task of choosing the appropriate burnable absorber it is of interest to search for nuclides, daughter nuclides of which can have some favourable effect on the development of a self-supporting chain reaction. In the work [8, 9] it is suggested to use ^{240}Pu as a burnable material, a method of enrichment of fuel with various initial content of even-numbered isotopes with proposed by the authors in the work [10]. In our case, the presence of isotope ^{240}Pu in a fuel composition results in significant accumulation of ^{241}Pu, which can be well fissioned by epithermal neutrons, and its nuclear concentration $^{241}N(t)$ exceeds the nuclear concentration of ^{233}U by more than 2.5 times during the whole fuel campaign (see fig. 3). The computation showed that ^{241}Pu and ^{233}U specify the dependence type of $k_{eff}(t)$ and provide steady reactor operation for 8.5 years.

It the loading option discussed above the initial reactivity excess is compensated by the isotope ^{240}Pu: its content (wt. %) in the fuel increased from 5.4 up to 9 %; the content of ^{239}Pu decreased by 3.6%, correspondingly. The use of ^{240}Pu in HTGRU does not lead to any undesired increase of reactivity in the middle of the fuel cycle as it happens in the core of PWR [8, 9]. Using ^{240}Pu $\rho_{initial}$ was reduced from 19.35 to 11.51%, and $(d\rho/dt)$ decreased by 1.51 times. According to the concept suggested by the authors in the works [4], the core of HGTRU, the axial part of which is substituted by a long magnetic trap with high-temperature plasma, starts up from the subcritical state ($k_{eff} = 0.95$). To transfer the core into the subcritical state it is enough to cover the side surface of fuel pellets with the 13 µm thick layer of ZrB_2.

2.3. Computational model of the axial zone of the core. Specific neutron yield of plasma thermonuclear source

The schematic drawing of the facility intended to study the fuel properties is presented in Figure 4. The axial zone of the core is substituted by the cylindrical vacuum chamber that contains high-temperature plasma generating high-energy neutrons because of D-D or/and D-T thermonuclear reactions. To inject deuterium beam in the plasma, a special chamber is attached to this cylindrical chamber. A magnetic field in these two jointed vacuum chambers containing the plasma provides heat isolation of this plasma from the chamber walls in a radial direction. Heat isolation of plasma along the magnetic field lines is provided by the multimirror sections of the magnetic field, which are adjacent to ends of the two jointed chambers with high-temperature plasma. The total length of the two chambers with high-temperature plasma together with two adjacent regions with the corrugated magnetic field is about 12 meters. The chamber for thermonuclear neutrons generation in the axial zone of the core [1] has the required length that is 3 meters.

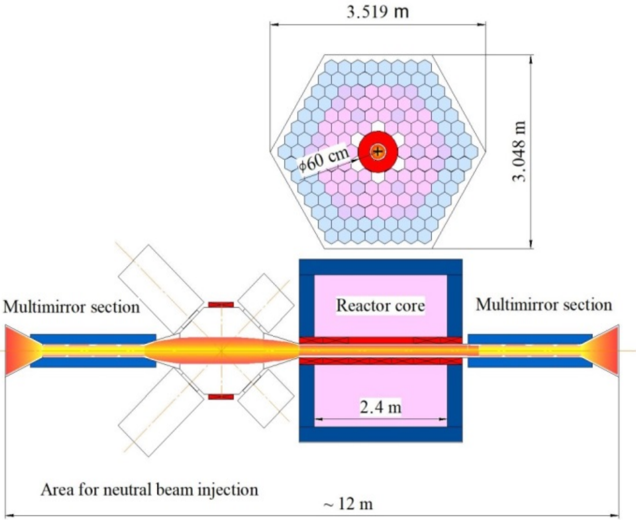


Figure 4. Schematic of the facility to study the fuel properties in long time operation of the reactor core.

Distribution of the magnetic field induction along the axis of the long (12-meters) plasma magnetic trap was chosen from considerations of the maximum neutron yield in the core plasma region upon condition of good homogeneity of radial neutron flow [5].

The distribution of the specific neutron yield per linear centimeter of the plasma column in case of D-D reaction along Z-axis calculated in the work [5] is plotted in Figure 5.

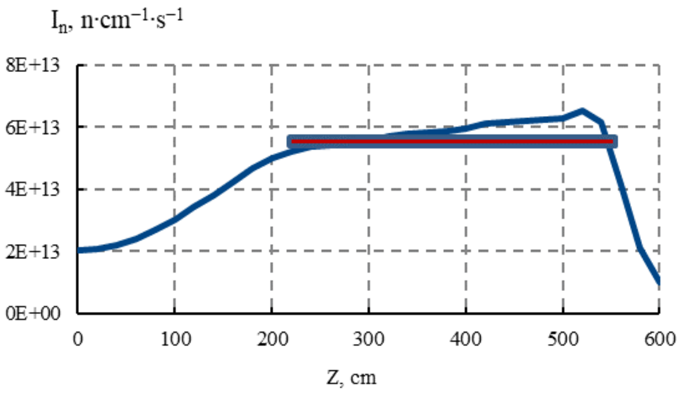


Figure 5. Distribution along Z-axis the specific neutron yield per linear centimeter of the plasma column I_n for the case of D-D fusion reaction. A red bar marks the part of the column that is situated inside of the reactor core.

A red bar located on the interval of z-coordinates from 200 cm to 500 cm marks the part of the column that is situated inside of the reactor core. One can see in Figure 5 that for chosen parameters of the magnetic field and the plasma column, the specific neutron yield changes insignificantly in the plasma column part that is placed in the core. Nevertheless, non-homogeneity of plasma distribution along Z-axis may be taken into consideration in our simulations. As examples of possibility to analyze various plasma distributions along Z-axis, the specific neutron yield of plasma neutron sources with cylindrical and conical shape of the plasma column was performed.

In numerical simulation, it was considered that plasma column is a volume source of monoenergetic neutrons with isotropic velocities distribution in the point of their origination. Neutrons are generated uniformly in the total volume of plasma and the neutrons energy is 2.45 MeV (for D-D reaction) or 14.1 MeV (for D-T reaction).

In accordance to results of the paper [5], when the density of high energy sloshing ions in the column of deuterium plasma 1.5×10^{14} cm^{-3}, the specific neutron yield in DD-reaction is 6.00×10^{13} n·s^{-1}·cm^{-1} (specific emission of neutrons from a volume unit is 1.76×10^{12} n·s^{-1}·cm^{-3}).

We calculated a spatial neutron flux distribution $\varphi_{Zn}(Z)$ on the external surface of the plasma neutron source with respect to axis Z for both two cases: 50 % T with 50 % D and 100 % D at the same specific neutron yield 6.00×10^{13} n·s^{-1}·cm^{-1}. We analyzed the neutron flux distribution for cylindrical and conical shapes of the plasma column. The diameter of the cylindrical plasma column was 17 cm. For conical shape of the column, its diameter was decreased from 17 cm down to 13 cm in the length 300 cm. The external surface of a cooper winding with the thickness 5 cm covering the vacuum chamber for the plasma column was the border between the plasma neutron source and the reactor core. Taking into account the thickness of the winding, the external surface of the plasma neutron source had the diameter varied from 22 cm to 18 cm for the conical shape of the plasma column and no varied 22 cm for the cylindrical one. The temperature of all materials used for the construction of the vacuum chamber and the solenoid was chosen at the level of 0.05 eV. The neutron flux $\varphi_{Sn}(E)$ on the external surface was generated by usage the program in the energy group structure ABBN-78 (of 28 energy neutron groups). Spatial neutron flux distribution $\varphi_n(Z)$ with respect to axis Z of the facility and its radius Y was obtained by placing the computational grid with the size of 1000 to 1000 cells along the height Z ($\varphi_{Zn}(Z)$) and radius Y ($\varphi_{Yn}(Z)$) of the model. The classic method of Monte-Carlo was used to solve the task in the frame of MCNP5 (ENDF/B-VII.0) [11], Serpent 1.1.7 (ENDF/B-VII.0) [12] and PRIZMA (ENDF/B-VII.I) [13] application.

3. Calculation data and conclusion

In accordance calculations, the total escape of D-D neutrons from the facility is 80.4%, neutrons escape from the side and end surface is 66.5 (φ_{Zn} = $6.67{\times}10^{12}$ n·s^{-1}·cm^{-2}) and 13.9 % ($\varphi_{Yn} = 1.39{\times}10^{13}$ n·s^{-1}·cm^{-2}), correspondingly.

The total escape of D-T neutrons from the facility is 98.5%, neutron escape from the side and end surface is 83.6 ($\varphi_{Zn} = 8.39{\times}10^{12}$ n·s^{-1}·cm^{-2}) and 14.9 % ($\varphi_{Yn} = 1.49{\times}10^{13}$ n·s^{-1}·cm^{-2}), correspondingly. It should also be mentioned that thorium in the first row of the assembly transmutates into ^{231}Pa under the influence by D-T-neutrons form the 26th (that is, for neutrons in the energy range E_n from 6.5 to 10.5 MeV), 27th (for neutrons in the energy range E_n from 10.5 to 14 MeV) and 28th (for neutrons in the energy range E_n from 14 to 14.5 MeV) groups due to (n, xn)-reactions. Therefore, the core provides extra neutrons and fissionable isotope ^{233}U.

In accordance calculations, neutrons escaping from the side surface of the facility can be divided in four main group with the different energy E_n. Percentage of these groups of the neutrons is the following: 2.03% – E_n from zero to 4 eV (thermal neutrons), 30.19% – E_n from 4 eV to 183 keV (epithermal neutrons), 50.11 % – E_n from 183 keV to 10.5 MeV and 17.69 % – E_n from 10.5 to 14.5 MeV (fast neutrons). In the reactor core [1,3] there are 9.41 % of thermal neutrons, 35.57 % of epithermal and 55.02 % of fast neutrons, correspondingly.

To achieve possibility of exactly correct usage of neutron emission from the plasma source for studying neutronic and thermal-physical properties of thorium-plutonium fuel in the core of HGTRU [1,3] we have to achieve the specific neutron yield on the level $I_n = 1.8{\times}10^{14}$ n·s^{-1}·cm^{-1}. This increase in the neutron yield will be realized in case of replacing some quantity of deuterium by tritium in the plasma column without changing any other parameters of the plasma.

Computer simulation of neutronic processes in the core of the high-temperature gas-cooled thorium reactor for 30 different options of core loadings was performed in the frame of our work. The quantity of fuel compact dispersion phase and the composition of fissionable nuclide were selected. Production of extra neutrons by means of thermonuclear reactions occurring in high-temperature plasma and of reactions of (n, xn) type was analyzed. Possibility to apply plasma D-T-source of neutrons to modify the near-axial region of HGTRU core was demonstrated. It was shown that the developed models and computer codes for description of the core and the thermonuclear neutrons source allow turning to full-scale study aimed at creation of a thorium subcritical assembly with supply of extra neutrons from thermonuclear plasma at its confinement in the long magnetic trap.

These studies are of undoubted practical interest, since they are necessary to substantiate the safety of operation of such hybrid systems. The research results will allow optimizing the active zone of the hybrid system with leveling the resulting offset radial and axial energy release distributions. Results of our study will be the basis for the development of new and improvement of existing methods of criticality control in related systems such as “pulsed neutron source - subcritical fuel assembly”.

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