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HYBRID FISSION-FUSION SYSTEMS AND STATISTICAL SIMULATION OF PARTICLE TRANSPORT

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The article presents comparative analysis of characteristics of different hybrid systems: ADS systems, a neutron source system based on TOKAMAK and systems with a neutron source based on the open plasma traps. The focus is on the development of the latter ones, which are noted for technical simplicity, low-cost source and high utilization of neutrons. It is shown that the developed system of codes for the simulation of the plasma neutron source based on gas-dynamic trap (GDT), neutron transport, and isotope kinetics allows calculations to optimize various parameters of hybrid system, and obtain necessary characteristics to analyze the technical and economic efficiency of such systems.

Key words: hybrid fission-fusion systems, statistical simulation, plasma dynamics, neutron emission, neutron transport process.

Introduction

In recent years, of particular interest is the possibility to create hybrid nuclear fusion systems (fission-fusion systems), which are the principal element of a closed nuclear fuel cycle (CNFC). These systems can be used to produce nuclear fuel from natural uranium or thorium, as well as to develop energy systems of average and low power (up to 500 MW electrical power output). Furthermore, hybrid systems are stations for processing and post-combustion of long-lived radioactive waste (primarily minor actinides). The relevance of such an effective and safe solution of the problem is caused by the fact that the rapid accumulation of nuclear fuel waste is a problem which could be one of the main reasons to abandon nuclear power, which is already the case in some developed countries. Currently various types of hybrid systems are in the process of development.

A number of foreign science teams develop Accelerator Driven Systems (ADS), which use reactions caused by accelerated electron beam of high energy particles to generate primary neutrons, Their advantage is the successful development of required technologies (SNS, OMEGA, JAERI EFIT-ADS) [1]. The ultimate localization and high power consumption of neutron source, low energy neutrons are irremediable defects to date.

Neutron source systems based on TOKAMAK (FDS) [2] have a high neutron yield. Parameters of currently available experimental facilities are close to the required technology. They have a fairly high coefficient of energy self-sufficiency Q. However, the technical complexity and high cost of the neutron source (estimated to be 1-3 billion dollars), electric corrosion of materials, low geometric efficiency of neutrons, high thermal and neutron load on the first wall, somewhat reduce their competitiveness.

Currently neutron source systems based on open traps (e.g., plasma gas dynamic traps – GDT are actively developed [3]). One of the advantages of such systems (as well as FDS) over ADS is that 14 MeV thermonuclear neutrons have much higher energy than neutrons generated at the source of accelerating type during spallation reaction. This provides more opportunities to increase the efficiency of neutrons generation in the reactor core using the appropriate coolants (e.g., lead) due to reactions (n, 2n) and even (n, 3n). Moreover, 14 MeV neutrons increase the efficiency of fission of actinides and plutonium, which reactions with low-energy neutrons are of small cross-sections.

Technical simplicity and low cost of the source, high neutron efficiency, possibility to choose a source spatial configuration distinguish hybrid systems based on open traps compared to systems with neutron source based on TOKAMAK.

There are still problems that developers have to solve in the foreseeable future – low coefficient of energy self-sufficiency Q of existing experimental facilities, a large circulation of tritium in the plant, a high neutron load on the first wall. The consumption of expensive tritium is low, but owing to fore-and-aft losses of tritium from the plant it is necessary to recuperate it and return to the neutrals source regularly.

The paper presents the results of numerical experiments aimed at determining the optimal configuration of hybrid system elements with a neutron source based on an open trap. Optimal configuration maintains maximum energy efficiency of the neutrons source and neutron efficiency of the source in blanket. Simulation programs of neutron processes in the blanket (NMC code [4]) and those of dynamics of plasma parameters in a gas-dynamic open trap (GENSYS code) are used in the calculation.

Scheme of subcritical hybrid system

Schematic diagram of subcritical hybrid system with a driver based on GDT is shown in Figure 1. This trap is a long axially symmetric probkotron for confinement of two-component non-equilibrium plasma [5]. The first component is warm collisional plasma confined in a gas-dynamic mode. The second one consists of fast ions that appear due to the capture of heat atomic beams on the warm component. The path length of fast ions significantly exceeds the size of the trap, so they are confined as in the classic "weakly collisional" probkotron. Since fast ions move in collisionless mode, and injection takes place at an acute angle to the magnetic field (pitch angle), oscillation of fast population particles (bounce-vibration) with adiabatically stable magnetic moment of particles takes place between the magnetic mirrors of the trap This effect results in the so-called stop points at which longitudinal velocity of particles having maximum density, is equal to zero. Since neutron emission takes place mainly due to fast component ions, the maximum emission power is namely at the stop points.

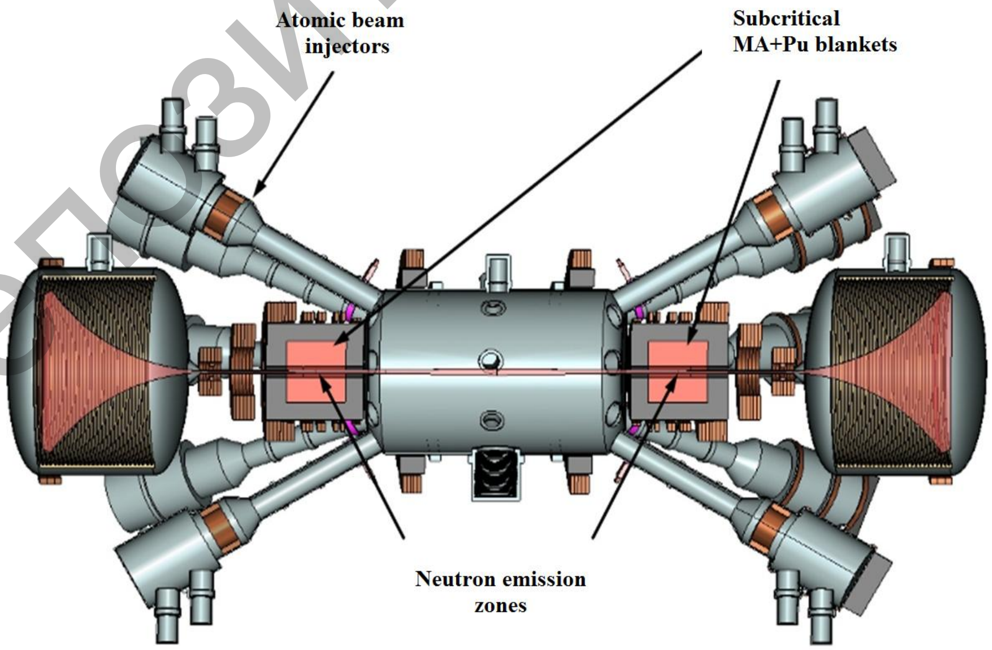

Fig.1. Schematic diagram of the hybrid system.

Around the stop points we suggest to locate fuel blankets. As a first step to carry out optimization, it was proposed to use a template based on a homogenized fuel blanket of EFIT system [6] (the blanket model is shown in Fig. 2). As the fuel area filler nitrides of minor actinides and plutonium were used. In this system, the coolant is a lead-bismuth eutectic and the reflector is a homogenized mixture of structure elements and coolant. When making neutron and physical calculations the source was presented as a string, and it was assumed that neutron emission power is constant along the axis of the fuel area. This simplification is valid if the size of fuel assembly is relatively small as compared to that of the entire plant.

Calculation of plasma dynamics and level of neutron emission of a source

To calculate the parameters of the neutron source based on the GDT a zero-dimensional code for calculating the plasma dynamics GENESYS (GENeral Evaluation SYStem) was developed. Zero-dimensionality of the code implies that the duration of the described processes is longer than the time of the spatial alignment of temperature and density of the warm plasma.

Dynamics of the hot component in GENESYS is described by the Fokker-Planck equation with consideration of a separate distribution function of deuterium and tritium particles. Distribution parameters of fast ions are energy and time. The angular distribution is considered a combination of Gaussian profiles with a time variable amplitude (the number of particles is $N(t, E)$) and angular width (pitch angle range is $\Delta\theta$ (t, E)) on the condition of "zeroing" of the amplitude at the boundaries of the loss cone. It is assumed that homogeneous energy atoms with a small angular range of pitch angles get into plasma during the injection.

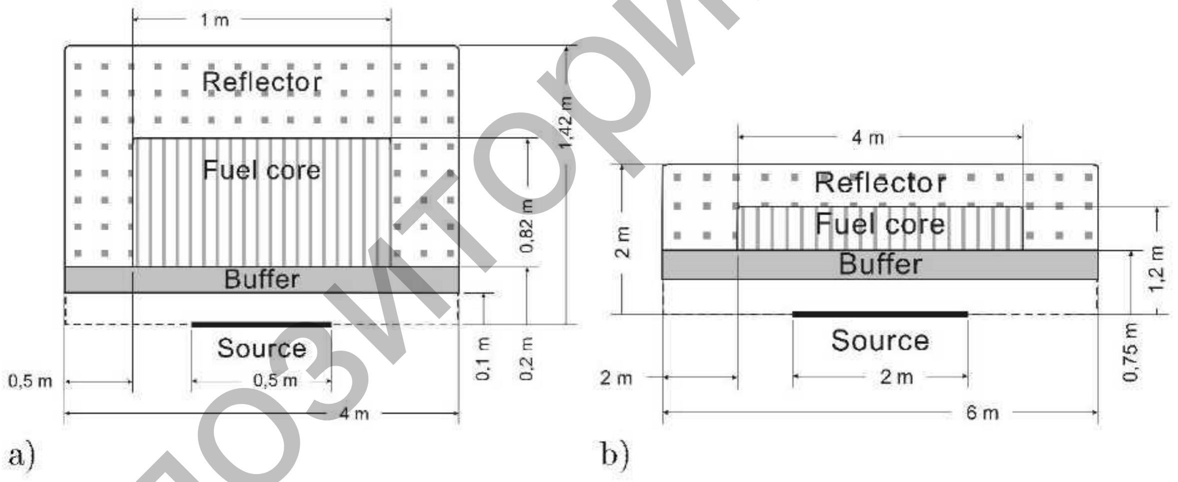

Fig.2. The scheme of subcritical blankets: a) the original version, b) a modified version.

When calculating the parameters of the warm plasma, we considered energy exchange processes with the hot plasma due to Coulomb collisions, the longitudinal loss of particles and energy through the magnetic mirror in the condition of the collisionless regime of flow and cold gas feeding into the chamber to maintain the density level of warm plasma. In order to achieve greater accuracy in computing, the model provides separation of plasma to peripheral and central areas, the latter of which coincides with the zone of fast particles motion between stop points. This separation makes it possible to simulate the displacement of the ions of the warm component off the center of the trap at a high density of fast particles. It is necessary to take into account this effect when simulating the compression of lateral density distribution of fast ions.

GENESYS code is also equipped with an additional module for calculating neutron emission per unit length. This calculation assumes a collisionless motion of fast particles along the axis of the trap (i.e., the code block is static, and the duration of the considered processes is considerably shorter than that of the distribution function change due to Coulomb interactions). The result of the calculation is the neutron emission power at a specified point along the axis. The simulation is performed using the Monte Carlo method. To verify the developed code, we compared the results of its work with the data of a number of experiments at GDT, presented in [7, 8]. In particular, Figure 3 shows the results of calculating the neutron emission power in the experiment with D-D mixture. As one can see from the graph, the results of the code show satisfactory convergence with experimental data. The accordance of calculation and experiment provided evidence for the parameters of energy and angular distribution of fast particles, and the electron temperature as well.

Simulation of neutron transport process

Neutron-physical processes in the blanket of subcritical systems were simulated using the *NMC* (Neutral Monte-Carlo) code [4]. One of the main advantages of the code is its implementation using an object-oriented approach that makes it possible to employ the code as a set of objects that perform various stages of simulation with a consistent interface and interchangeable within their own functionality. The code enables a user to extend its functionality by implementing own software.

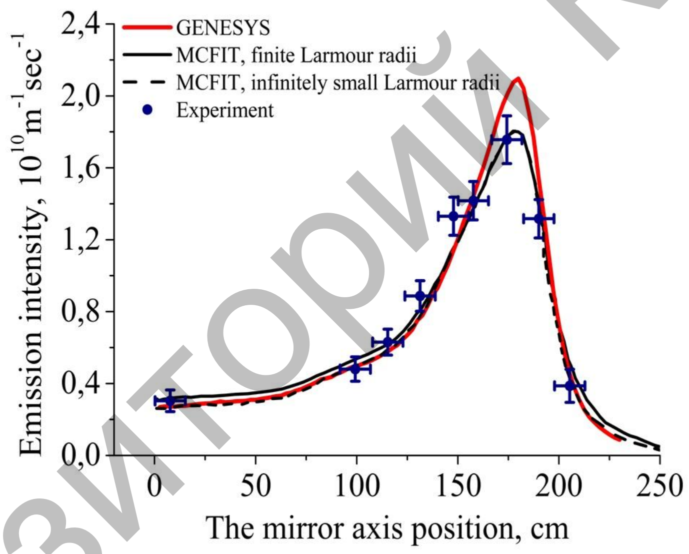

Fig. 3: The dependence of neutron emission intensity on the coordinate along the axis of the trap.

NMC is a static code using the Monte Carlo method to calculate the neutron parameters of the system under consideration. Thus, the distribution of the neutron field of a given system is determined by solving the transport equation by means of direct simulation of particle trajectories. We used a grouped approximation in calculations. Preparation of group cross-sections was carried out using PREPRO processing code. The source data were taken from the library of evaluated neutron data ENDF/B-VII. The code functioning was thoroughly verified in a number of test builds of critical experiments libraries [9].

Maximization of the integrated neutron emission source

Optimization of parameters of the emission zone was carried out for one of the latest published versions of the neutron source prototype [10].

The optimization of neutron emission power implied change in the magnetic field profile of the source, while keeping unchanged the rest of the input parameters. The value of the magnetic field in the mirror was fixed and the area length with the magnetic field close to the value at the stop point varied. It should be noted that such magnetic field configuration is different from the one that should be created by the magnetic system in the experimental plant, since the GENESYS model does not take into account the impact of fast particles on the magnetic field profile and paraxial field requirement.

When optimizing the emission zone we considered two essentially different versions of the source, conditionally designated as "optimistic" and "pessimistic" ones. The optimistic version assumes the effect of the cross-pinching density of fast particles. In the pessimistic case, the density distribution of fast particles and that of the background plasma coincided. As a result, we calculated the dependence of the neutron emission power on the length of the test area for the optimistic and pessimistic versions of the neutron source in cases of one- and two-band configurations. The results of the calculations are shown in Figure 4.

Note two important points resulting from the calculations. First, the cross-pinching of fast particles leads to approximately twofold increase in neutron emission in the test area. This effect is a direct consequence of fast ions density growth and reduction in their motion area. For a given magnetic field strength, we can neglect the effect of the finite Larmor radii of fast particles, neglected in the model GENESYS.

Another important effect is the optimum of the emission zone length. This effect is due to the fact that a high density of fast particles leads to increased concentration of electrons in the central part of the trap, and corresponding increase in electron drag. At the same time, decrease in fast particles density when extending the emission zone results in fewer thermonuclear interactions. As a result of the competition between two described processes there is an optimum neutron flux power at a certain value of the length of the emission zone.

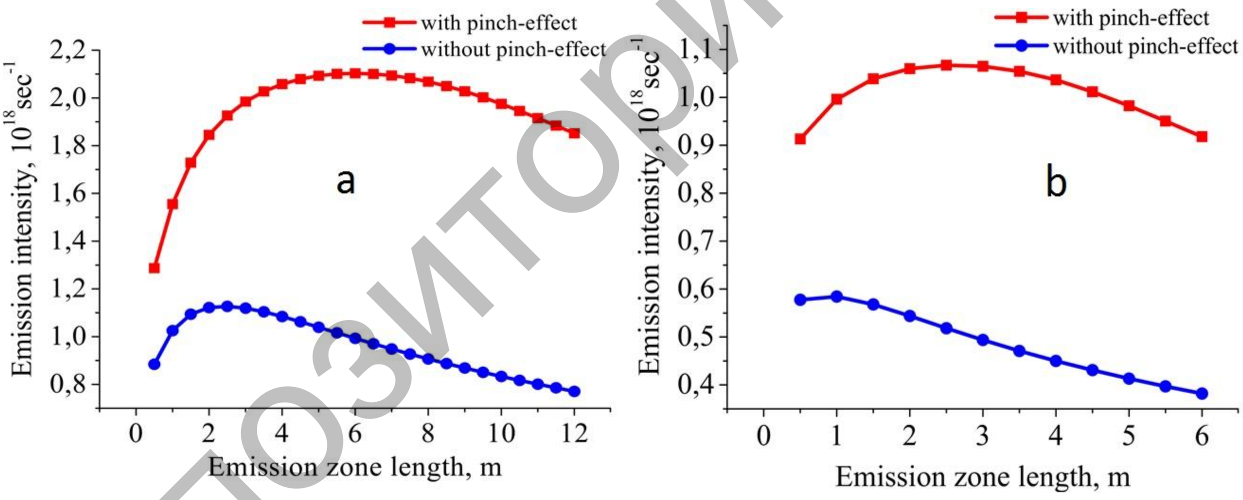

Fig. 4. The intensity of the emission for the single-band configuration source (s) and a source with a centrally symmetric magnetic field (b).

Table 1. The comparison results for the blankets presented in Fig. 2.

Parameter	Configuration A	Configuration B
The thickness of the buffer	0 cm	10 cm
Criticality, MCNP	0,95008	0,95817
Criticality, NMC	0,9534	0,9608
Multiplicativity, MCNP	34,75	44,38
Multiplicativity, NMC	37,1	51,3

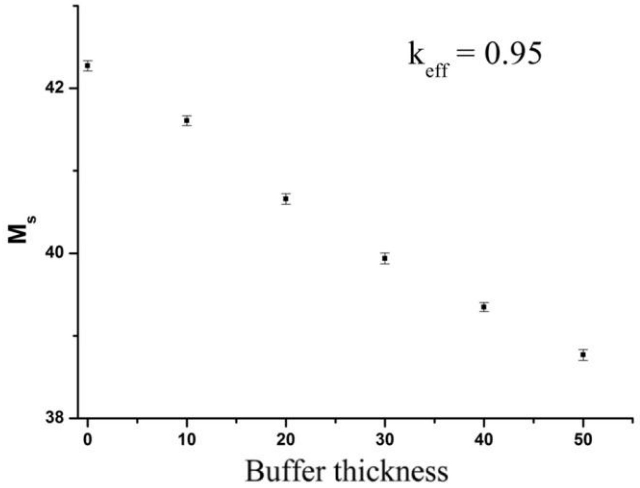

Fig. 5. The dependence of multiplicativity on the thickness of the buffer.

Effect of the thickness of the buffer on the reactor multiplicativity

Multiplication and critical factors for the fuel blanket described above in two different geometric configurations shown in Fig. 2 were calculated using the *NMC* code. The main purpose of the calculation was to optimize the multiplication factor by changing the thickness of the buffer zone at a fixed blanket criticality. The version presented in Figure 2-a was used for additional validation of the *NMC* code and comparison with the data previously described in [11]. The results of the calculations are shown in Table 1, where H is the thickness of the buffer, k_{eff} is the multiplication factor, M_{S} is multiplicativity.

For further calculations we used a modified version of the blanket shown in Figure 2-b. Its use is due to improved fuel exposure in relation to the neutron source and increased characteristic effective length of the source. The thickness of the buffer zone when modeling varied in the range of 0 to 50 cm, and the fuel area configured so that at a predetermined buffer thickness the criticality factor was k_{eff} =0.95. The results of calculation of multiplication factor for a fixed criticality are shown in Figure 5. The figure shows that growth of buffer zone thickness leads to a lower multiplication factor. This effect is caused by transfer of neutrons in the buffer to a lower energy range (of 1 MeV order) due to reactions (n, 2n) in lead. It causes the growth of ratio of capture cross-section to fission cross-section in minor actinides and corresponding reduction in the multiplication factor.

Conclusion

We carried out the comparative analysis of characteristics of different hybrid systems: 1) ADS systems; 2) systems with a neutron source based on TOKAMAK ; 3) systems with a neutrons source based on open plasma traps.

We determined the level of neutron emission for configuration of neutron source on the basis of the GDT, the parameters of which can be considered close to critical.

The neutron emission power has a flat maximum of the length of emitting area due to competition between the electron drag and fast particles density change processes. In the optimistic scenario of the two-band source configuration, its maximum power of is achieved at 2-3 meters length of the emission zone and is 1.14 10^{18}. In case of pessimistic scenario, the maximum is observed in small length area, as the critical density is achieved only at a significant reduction in the length of the motion area of fast particles. We calculated the dependence of the integrated coefficient of sub-critical hybrid system blanket on the thickness of the buffer zone, consisting of a lead-bismuth eutectic, this assembly being irradiated by neutrons with energy of 14.1 MeV. We found that the maximum value of the multiplication factor is achieved by removing the buffer zone from the fuel blanket.

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