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Research Article

Fusion-fission hybrid reactor facility: power profiling*

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Abstract

The current state of research in the field of nuclear and thermonuclear power aimed at creating power generation plants makes it possible to predict the further development of modern power industry in the direction hybrid reactor power plants. Such hybrid systems include a tokamak with reactor technologies, worked out in detail in Russia, and systems with an additional source of neutrons.

Power generation plants using tokamaks and accelerators with the required level of proton energy will be of exceptionally large size and power, which will postpone their construction on an industrial scale to the distant future. The ongoing research is aimed at the development of small generation and has the prospect of entering the field of energy use in a shorter period. The hybrid reactor facility under study consists of an axisymmetric assembly of fuel blocks of a high-temperature gas-cooled reactor and a linear plasma source of additional neutrons. The paper demonstrates the results of optimization plasma-physical, thermophysical and gas-dynamic studies, the purpose of which is to level the distortions of the power density field, which are formed in the volume of the multiplicating part of the facility due to the pulsed operation of the plasma source of D-T-neutrons. The studies on increasing the "brightness" of the source and modeling its operating modes were carried out using the DOL and PRIZMA programs. The thermophysical optimization and gas-dynamic calculations were performed using the verified SERPENT and FloEFD software codes. The calculations were made on a high-performance cluster of the Tomsk Polytechnic University.

Keywords

fusion-fission hybrid reactor facility, plasma source of D-T-neutrons, power profiling, temperature field

Introduction

The current state of research in the field of nuclear and thermonuclear power (Abderrahim et al. 2001, Moir et al. 2012, Shmelev et al. 2015, Arzhannikov et al. 2016, 2019, 2020, Knaster et al. 2016, Wu 2016, Ananyev et al. 2020, Prikhodko and Arzhannikov 2020, Bedenko et al. 2021a, b, Gudowski et al. 2021, Krasilnikov et al. 2021, Salvatores et al. 2021, Atomnaya Energiya 2.0. Rosatom Plans to Expand Research in the Field of Thermonuclear and Plasma

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Technologies 2022, AtomInfo 2022). The Belgian Nuclear Center SCK-CEN Provided Information on a Course of Construction the New Isotope MYRRHA Reactor, 2022) aimed at creating power generation plants makes it possible to predict the further development of modern power industry in the direction hybrid reactor power plants. Therefore, in February 2022, at the Seminar on Controlled Fusion and Plasma Technologies, Rosatom State Corporation announced tokamak-based systems using fusion neutron sources (Krasilnikov et al. 2021, Atomnaya Energiya 2.0. Rosatom Plans to Expand Research in the Field of Thermonuclear and Plasma Technologies 2022).

Such systems include the tokamak with reactor technologies (TRT) worked out in detail in Russia (Krasilnikov et al. 2021, Atomnaya Energiya 2.0. Rosatom Plans to Expand Research in the Field of Thermonuclear and Plasma Technologies 2022), which is planned to be put into operation at at the Troitsk Institute for Innovation and Thermonuclear Research (TRINITI) by 2030 (Atomnaya Energiya 2.0. Rosatom Plans to Expand Research in the Field of Thermonuclear and Plasma Technologies 2022).

Another concept includes systems with an additional source of neutrons produced by bombarding a target with a beam of high-energy protons (Abderrahim et al. 2001, Knaster et al. 2016, Prikhodko and Arzhannikov 2020, AtomInfo. The Belgian Nuclear Center SCK-CEN Provided Information on a Course of Construction the New Isotope MYRRHA Reactor 2022).

Power generation plants using tokamaks and accelerators with the required level of proton energy will be of exceptionally large size and gigantic power, which will postpone their construction on an industrial scale to the distant future.

The ongoing research (Arzhannikov et al. 2016, 2019, 2020, Prikhodko and Arzhannikov 2020, Bedenko et al. 2021a, b) is aimed at the development of small generation and has the prospect of entering the field of energy use in a shorter period. The facility under study is a hybrid reactor, the core (blanket) of which consists of an assembly of prismatic graphite blocks (Shamanin et al. 2015, Bedenko et al. 2019a) and an extended plasma source of additional D-T-neutrons of a linear configuration (Anikeev et al. 2015, Yurov et al. 2016).

The use of such a source makes it possible to transfer the entire system to a subcritical operating mode and thereby dramatically increase the level of its nuclear safety, as well as to ensure more efficient use of fuel by changing the "hardness" of the neutron energy spectrum.

It was shown in Prikhodko and Arzhannikov 2020, Bedenko et al. 2021b, that when a neutron flux with an intensity of 2.56×10^{17} n×s^{-1} enters the multiplicating part of the facility from the plasma neutron source (PNS), the flux will increase to approximately 1 $\times10^{20}$ n×s^{-1}. In the pulse-periodic mode of operation of the PNS, distortions of the power density field occur. As a result, a temperature field gradient is created, which requires a study of the dynamics of the formation of spatial power density.

The paper demonstrates optimization plasma-physical, thermophysical studies and the results of gas-dynamic modeling, the purpose of which is to level the resulting distortions of the radial and axial power density fields, which are formed in the blanket due to the pulsed mode of operation of the PNS.

Computational methods

The facility under study (Fig. 1) consists of a plasma source of D-T neutrons and a blanket part. The blanket part is based on the concept of the core of a multi-purpose high-temperature gas-cooled reactor facility of low power (Shamanin et al. 2015, Bedenko et al. 2019a) with a near-axial region modified for the plasma neutron source (Arzhannikov et al. 2016). The modified core (Yurov et al. 2016, Bedenko et al. 2021b) is a blanket composed of prismatic graphite fuel and non-fuel blocks containing dispersed ($^{\text{weapons-grade(wg)}}$Pu–Th)$O_2$-fuel in cylindrical channels (Bedenko et al. 2019a).

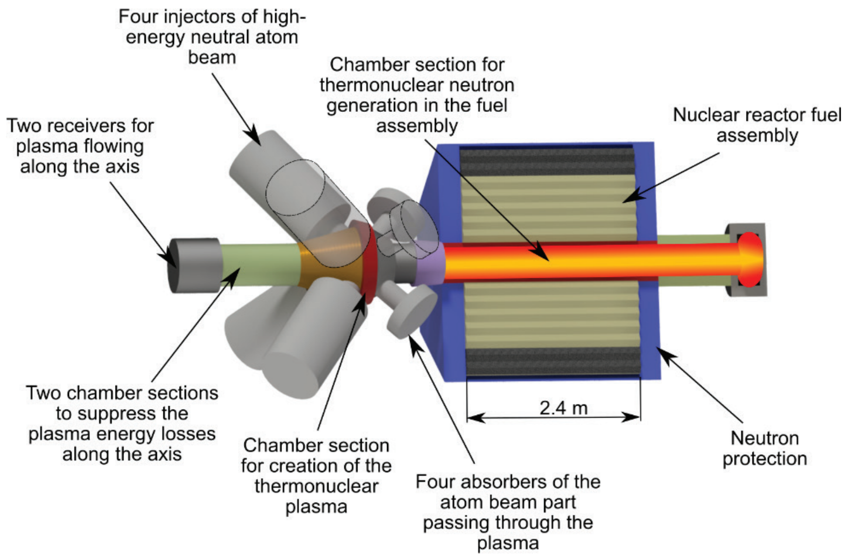

Figure 1. Conceptual design of the fusion-fission hybrid reactor facility (Arzhannikov et al. 2016, Bedenko et al. 2021b).

The PNS is a plasma vacuum chamber with a linear configuration (Anikeev et al. 2015, Yurov et al. 2016), in which the high-temperature plasma is contained by a magnetic field. This area is limited by magnetic mirrors, which in Fig. 2 are located at coordinates $y_1 = -5$ m and $y_2 = 10$ m.

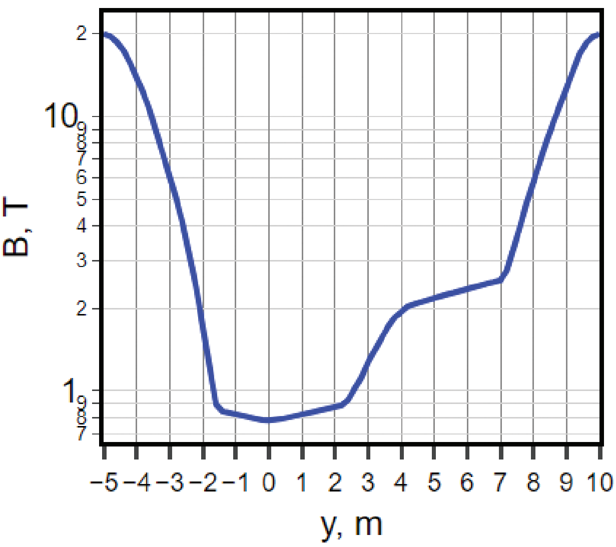

Figure 2. Magnetic field induction $B(y)$ as a function of the y coordinate along the facility axis.

This area of the vacuum chamber can be divided into two parts. The first one is designed to accommodate atomic beam injectors that deliver energy into the plasma. This part of the chamber has a large diameter and serves as the main volume where the plasma component with warm ions is kept. The magnetic field induction in this part of the chamber is practically independent of the y coordinate in the range of -1.8 m $< y < 2.2$ m. The second part of the chamber (see coordinates 4 m $< y <$ 7 m in Fig. 2) is located inside subcritical blanket. In this three-meter part of the chamber, D-T-neutrons are generated, which enter the blanket. The magnetic field builds up slowly here, providing a uniform neutron production profile. Based on the plasma parameters achieved in the computational experiments (see the field profile $B(y)$ in Fig. 2), we used the DOL code (Anikeev et al. 2015) to simulate the generation of D-T-neutrons with the maximum yield of I_n into the blanket part at a level of about $1\times10^{18} - 1\times10^{19}$ n$\times$s^{-1}.

The longitudinal profile of the yield of these neutrons $I_n(y)$ was used to determine the peak values of the energy release in the problem of thermophysical optimization of the multiplicating part of the facility.

It should be noted that in the “PNS – blanket” configuration under study, the plasma column is formed in a repetitively pulsed mode, and the “wave” of fission of $(^{wg}Pu–Th)O_2$-fuel nuclei propagates from the axial region over the entire multiplicating part in proportion to the time with the PNS operation. The study of the dynamics of fission of fuel nuclei was carried out using the PRIZMA code (Kandiev et al. 2015) in combination with subprograms specially developed by the employees of the All-Russian Scientific Research Institute of Technical Physics (RFNC-VNIITF).

During the simulation, the repetitively pulsed and stationary modes of operation of the facility were studied. The simulation results (Fig. 3, Layer 1 is included in the computational region of the second row of fuel blocks and Layer 50 is included in the computational region of the fourth (peripheral) row of fuel blocks) have shown that a reduction in the pulse duration with the same pulse repetition period leads to an increase in the number $S(t)$ of fissile fuel nuclei per unit time, while with an increase in the pulse duration at the same value of neutrons in the pulse, the opposite effect is observed. It can also be seen in the figure (see Lines 2 and 3) that under conditions where the pulse duration is on the order of 1 ms, the effect of neutron emission nonstationarity is weakly manifested in all the fuel blocks. This result allows us to consider the pulsed source as quasi-stationary and use the SERPENT 2.1.31 reactor code for the purposes of thermophysical optimization. The simulation used evaluated nuclear data in ACE format converted from ENDF-B/VII.1 with an additional library of nuclear data in the field of neutron thermalization for graphite.

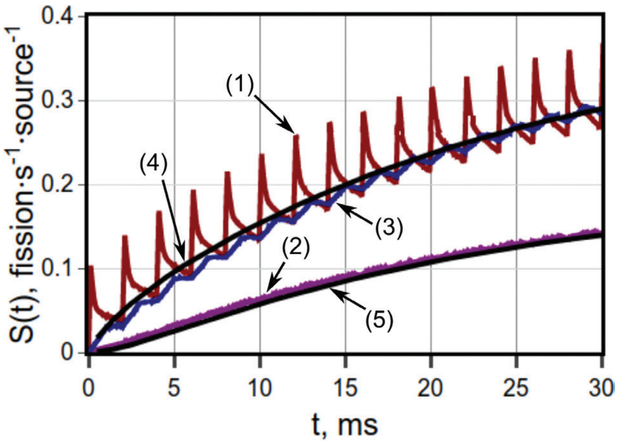

Figure 3. Comparison of the rate of nuclear fission in the blanket of the facility under repetitively pulsed (Lines 1–3) and stationary (Lines 4, 5) operating modes of the PNS: (1) Layer 1 [t = 0.1 ms, T = 2 ms]; (2) Layer 50 [t = 0.1 ms, T = 2 ms]; (3) Layer 1 [t = 1 ms, T = 2 ms]; (4) Layer 1; (5) Layer 50.

Gas-dynamic modeling was performed by the finite element method in a conjugate formulation: both heats transfer due to heat conduction and convective heat transfer were taken into account. The blanket configuration is shown in a fig. 2 from Bedenko et al. 2021b. The standard FloEFD mesh generator from the Siemens Simcenter FloEFD 2020.2 program code was used to build the grid model. The two-wall near-wall functions implemented in FloEFD were used to simulate the turbulence of the helium flow.

This approach makes it possible, depending on the parameters of the flow and the size of the grid in the near-wall region, to calculate the boundary layer in two versions (models of a "thin" and "thick" boundary layer) with satisfactory accuracy. For effective cooling of the blanket and PNS, a series of optimization parametric calculations was carried out, in which the variables were the coolant parameters (temperature, pressure, and velocity), and the target parameters were the temperature extremes of the source and blanket materials. The parameters of helium, the thermophysical properties of helium, structural materials, and fuel were chosen with due account of the experience in designing and operating the HTTR reactor (Bess and Fujimoto 2014), as well as the results of studies presented in Bedenko et al. 2019b.

Results and discussion

The thermophysical optimization of the blanket was performed by profiling the power density along the radius of the multiplicating part by changing the content of the Pu fraction. The fuel blocks from the second to the fourth row (see specifiers 1–4 in fig. 2a from Bedenko et al. 2021b) are loaded with fuel with the volume fraction of the dispersed phase w_{pf} = 7, 13, 17, and 21% (base value = 17%). With this load (see the right 1/12th part of the computational model segment in Fig. 4a), the radial power density profile becomes more uniform, while the peak power density values are reduced by 19% compared to the non-profiled blanket.

The results of numerical gas-dynamic modeling presented in Fig. 4 show that the temperature of the first non-fuel and the second row of fuel blocks is reduced to the values required for the PNS normal operation.

The most energy-intensive area of the blanket is located in the third row at a height of 2.56 m (Fig. 4b). The maximum temperature of the fuel and adjacent graphite here reaches 1335 °C and varies slightly along the radius, thus providing an almost uniform temperature profile of the third and fourth rows of the blocks. Note that the operating temperatures of dispersed (^{wg}Pu–Th)O_2-fuel and graphite should not exceed 1250 °C and 1300 °C, respectively (Bess and Fujimoto 2014, Bedenko et al. 2019b, Shaimerdenov et al. 2022). For the boundary conditions specified for the coolant (inlet helium velocity = 30 m×s^{-1}, temperature = 427 °C and pressure = 7 MPa), we see some excess of the temperature limit values. The chosen boundary conditions for the observed local overheating of the fuel and graphite make it possible to obtain the helium temperature (~ 800 °C at the outlet of the third-row fuel blocks) required for hydrogen production by methane steam reforming.

The power density of the third row can be reduced, for example, by changing the configuration of the flow area of the channels for the coolant, while maintaining the same flow area of the first, second and fourth rows of the blocks. Another option, which does not require additional neutron and thermophysical calculations, is to use local resistance, for example, in the form of so-called orifice gages or heat transfer intensifiers.

Conclusion

The paper demonstrates the results of optimization plasma-physical, thermophysical and gas-dynamic studies.

The results of studies performed at the Budker Institute of Nuclear Physics made it possible to improve plasma confinement (see Fig. 2) by suppressing longitudinal losses by multiple-mirror sections and to increase the "brightness" of the source from the level of 2.56×10^{17} n×s^{-1} to a record value of 2.77×10^{18} n×s^{-1}.

The operation of the plasma neutron source in conjunction with the blanket multiplicating part was simulated by our colleagues from the All-Russian Scientific Research Institute of Technical Physics (RFNC-VNIITF).

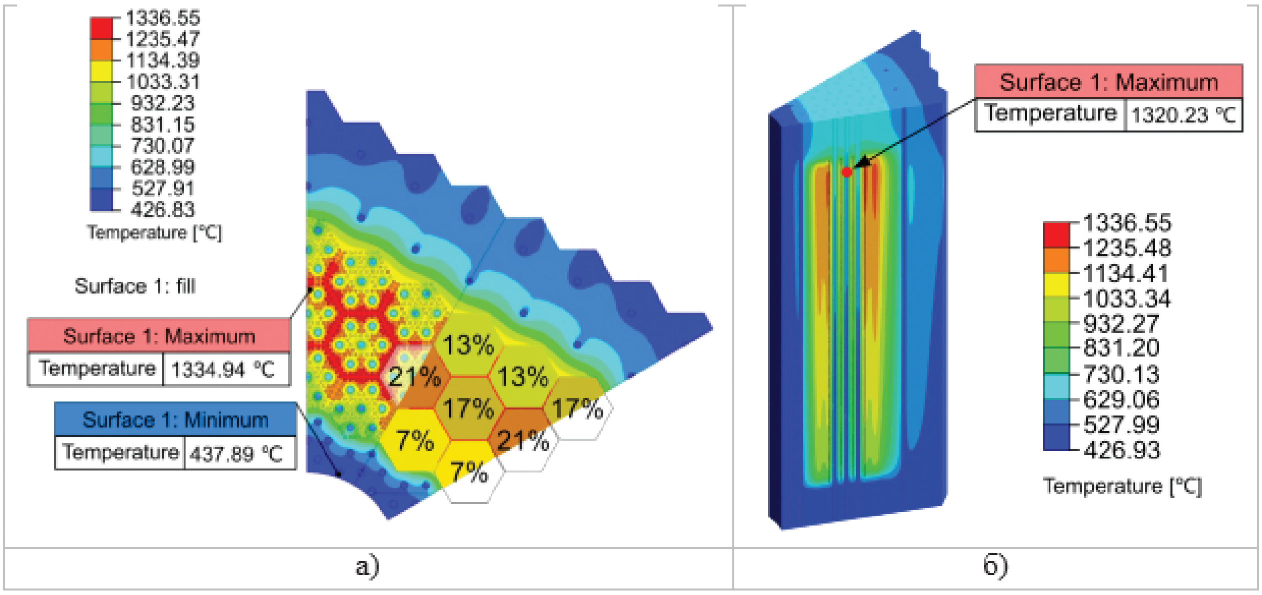

Figure 4. Loading scheme (**a**) and blanket temperature profile in the area with the maximum temperature (**b**).

In the course of this simulation, the dynamics of the process of fission of fuel nuclei and the formation of spatial power density in the pulsed and stationary operating modes of the facility were studied. The result obtained (see Fig. 3) made it possible to consider the pulsed source as a quasi-stationary one and perform thermophysical optimization (see Fig. 4a) in neutron studies using the SERPENT code.

The gas-dynamic studies of the facility blanket cooling conditions were carried out by Tomsk Polytechnic University in cooperation with OKB Gidropress. The results of parametric optimization calculations (see Fig. 4) showed a satisfactory agreement between the main operating parameters of the blanket and similar experimental parameters of reference HTGR plants.

The results of three years of work on the justification of the possible integration of the facility into the operating fleet of small generation allow us to proceed with the preliminary design of a safe subcritical hybrid reactor.

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