CONFERENCE PRE-PRINT

THE GLOBUS-3 PROJECT AS THE NEXT STEP IN THE RESEARCH PROGRAM ON SPHERICAL TOKAMAKS AT THE IOFFE INSTITUTE

V.B. MINAEV[1], V.K. GUSEV[1], YU.V. PETROV[1], N.V. SAKHAROV[1], N.N. BAKHAREV[1], I.M. BALACHENKOV[1], E.N. BONDARCHUK[2], A.V. BONDAR[2], P.A. BAGRYANSKY[3], F.V. CHERNYSHEV[1], V.V. DYACHENKO[1], O.G. FILATOV[2], V.YU. GORYAINOV[1], E.Z. GUSAKOV[1], E.G. KAVEEVA[4], A.A. KAVIN[2], I.V. KEDROV[2], N.A. KHROMOV[1], E.O. KISELEV[1], V.V. KOLMOGOROV[3], A.YU. KONIN[2], A.M. KUDRYAVTSEVA[2], K.A. KUKUSHKIN[4], G.S. KURSKIEV[1], A.N. LABUSOV[2], A.D. MELNIK[1], A.B. MINEEV[2], I.V. MIROSHNIKOV[1], A.N. NOVOKHATSKY[1], A.YU. POPOV[1], V.A. ROZHANSKY[4], P.B. SHCHEGOLEV[1], A.E. SHEVELEV[1], I.V. SHIKHOVTSEV[3], K.D. SHULYATIEV[1], O.M. SKREKEL[1], V.N. TANCHUK[2], A.YU. TELNOVA[1], E.E. TKACHENKO[1], V.A. TOKAREV[1], S.YU. TOLSTYAKOV[1], V.A. TROFIMOV[2], E.A. TYUKHMENEVA[1], V.I. VARFOLOMEEV[1], A.A. VORONOVA[2], A.YU. YASHIN[4], E.G. ZHILIN[5], N.S. ZHILTSOV[1]

[1]Ioffe Institute, St. Petersburg, Russian Federation
[2]JSC «NIIEFA», St. Petersburg, Russian Federation
[3]Budker Institute, Novosibirsk, Russian Federation
[4]Peter the Great St. Petersburg Polytechnic University, St. Petersburg, Russian Federation
[5]Ioffe Fusion Technology Ltd., St. Petersburg, Russian Federation

Email: Vladimir.Minaev@mail.ioffe.ru

Abstract

This paper presents the concept of the Globus-3 tokamak, which is the next stage of the research program on plasma confinement in toroidal magnetic facilities with a small aspect ratio at the Ioffe Institute. Distinguishing features of the facility include an increased toroidal magnetic field and major torus radius, and a plasma discharge duration exceeding the characteristic times for the formation of steady-state plasma parameter profiles. The design of the tokamak electromagnetic system with warm coils and the vacuum vessel is described, and a basic discharge scenario for the initial stage of experiments is considered. Methods for auxiliary heating and non-inductive plasma current drive planned for use at the Globus-3 tokamak are listed. Estimates of the plasma parameters obtained from modelling are presented.

1. INTRODUCTION

In 1999, the first Russian spherical tokamak Globus-M [1] ($R \approx 36$ cm, A ≈ 1.5) was put into operation at the Ioffe Institute. The facility features were a limited value of toroidal magnetic field of 0.5 T, a plasma column tightly inscribed in the vacuum vessel, and a large normalized Larmor radius ($\rho_i^* = \rho_i / a$). The main objective of the project was to experimentally demonstrate the theoretically predicted properties of magnetic configurations of plasma with a small aspect ratio [2]. The use of atomic beam injectors for plasma heating was not considered during the facility's construction. This plasma heating system was implemented after the facility's startup and provided a high power density of auxiliary heating. However, the heating efficiency was negatively affected by poor energy confinement and high level of losses of high-energy ions due to the low magnetic field [3]. As was discovered at the MAST [4] and NSTX [5] facilities and later confirmed at the Globus-M tokamak [6], the energy confinement time in a spherical tokamak, unlike a conventional one [7,8], has a strong dependence on the magnetic field. Taking these circumstances into account, the design of the electromagnetic system of the next tokamak, Globus-M2 [9,10], launched in 2018, was completely changed, which made it possible to significantly improve the technical parameters of the facility: to increase the toroidal magnetic field B_{tor} to 1 T and the plasma current I_p to 0.5 MA. The result was a significant improvement in energy confinement [11] and the achievement of a "hot ion" mode [12], in which the ion temperature exceeded 4 keV. The analysis showed that ion heat transport in this mode is well described by neoclassical theory [13]. Along with this, a number of problems were found that impose restrictions on the range of experimental parameters. The tokamak's small size complicates the injection of electromagnetic waves at resonant frequencies into the plasma. Placing antennas inside the vacuum vessel is impossible due to the lack of sufficient space between the plasma boundary and the wall. Antennas must be housed in the vessel's ports, which significantly limit the injected power. It should be noted that Globus is the only spherical tokamak in the world that successfully uses ICR heating, and experiments are planned to generate non-inductive plasma current using helicon waves. There are also limitations on the use of neutral beam injectors for plasma heating, including both beam size and, consequently, the injected power, as well as injection angles. Injecting a beam at an angle to the torus midplane (off-axis injection) is virtually impossible. The feasibility of developing the new Globus-3 facility is predetermined by two key considerations, the implementation of which will significantly improve the particle and energy confinement in the plasma. The first of these is the need to further increase the toroidal magnetic field while maintaining a small aspect ratio of the torus. Due to the limited diameter of the torus's internal bore, increasing the magnetic field and discharge pulse duration relative to the plasma current is a priority. If non-inductive start and plasma current drive methods will be successfully applied, these values can be further increased. The latter is associated with an increase in the vessel and plasma size. Increasing the plasma size, coupled with an increased magnetic field, will improve the confinement of fast ions generated by auxiliary ICR heating and neutral beam injection. Increasing the gap between the plasma boundary and the chamber wall on the low magnetic field side to 10-15 cm will allow for the placement of an antenna for the injection of resonant electromagnetic waves at various frequencies.

2. PROJECT CONCEPT AND DESIGN FEATURES OF THE GLOBUS-3 TOKAMAK

The Globus-3 tokamak [14,15] is the next step in the research program on plasma confinement in toroidal magnetic facilities with a small aspect ratio at the Ioffe Institute. Distinguishing features of the device include increased toroidal magnetic field and major torus radius, and a plasma discharge duration exceeding the characteristic times for the formation of steady-state plasma parameter profiles. The project proposes to continue using all the auxiliary heating and current drive systems (NBI, ICRH, LHCD) previously employed at the Globus-M2 tokamak, and, due to the increased magnetic field, to introduce an electron cyclotron heating (ECRH) system.

To reduce costs, it is planned to use the existing infrastructure during the initial phase of the facility's operation. The new tokamak will be placed in a 300-square-meter experimental hall on the site of the currently operating Globus-M2 tokamak. The tokamak's electromagnetic system and auxiliary heating systems will be powered by existing grid power supplies with a total capacity of 125 MVA. During the physical startup phase, the existing auxiliary plasma heating systems, diagnostic suite, water cooling, and vacuum pumping systems will be used.

During the pre-conceptual design stage, three electromagnetic system options were considered: with warm copper coils, with copper coils pre-cooled to liquid nitrogen temperature, and with superconducting coils. The last two were rejected due to the lack of technology readiness and problems with accommodating the facility in the existing experimental hall. The conducted analysis of the parameters and the development of the tokamak layout led to the following conservative version of the Globus-3 tokamak with a warm electromagnetic system: R_0 = 0.775 m, a = 0.440 m, A = 1.76, B_{T0} = 1.5 T, I_P = 0.8 MA, k_{95} = 1.8; discharge plateau duration $\Delta t_{plateau}$ = 2÷3 s. An extreme tokamak operation scenario with increased magnetic field of B_{T0} = 1.8 T and plasma current I_P = 2.0 MA with a reduced discharge duration was also considered. As a result, the overall dimensions of the Globus-3 facility are 3.8×3.8 m (transverse dimension × height), allowing the tokamak to be placed in the existing experimental hall.

In general, the layout of the electromagnetic system and vacuum vessel components of the Globus-3 tokamak (see Fig. 1) is conceptually similar to that of the Globus-M2 tokamak. The central solenoid is wound in two layers on the central assembly (column) of the toroidal magnetic field coil and provides a magnetic flux consumption of approximately 2 Wb. The toroidal magnetic field coil consists of 16 four-turn assemblies. The value of the magnetic field ripple on the outer plasma surface, Δ_{ripple}, is less than 0.4%. Coil overheating during a pulse does not exceed 40°C, and heat is removed from them during the pause between pulses (≤ 30 minutes). The gap between the plasma surface and the vacuum vessel wall is 40 mm on the high-field side and 135 mm on the low-field side.

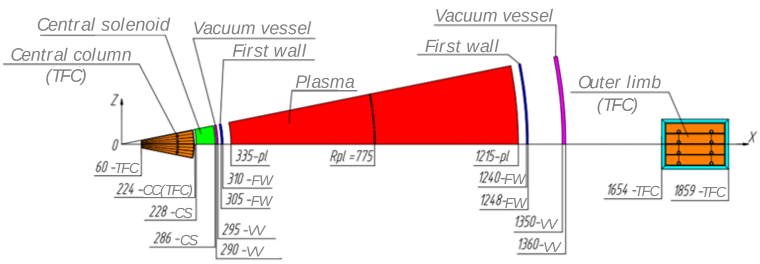


FIG. 1. Layout of the electromagnetic system and vacuum vessel of the Globus-3 tokamak along a large radius.

A vertical cross-section of the tokamak's electromagnetic system is shown in Fig. 2. The figure also shows plasma column configurations inscribed within the vacuum vessel with positive (left) and negative (right) triangularity, which can be implemented in Globus-3. It should be noted that, unlike the Globus-M/-M2 tokamak, the warm version of the Globus-3 facility requires the installation of passive stabilization coils inside the vacuum vessel. The coils are designed as two electrically open rings located in the upper and lower areas of the vacuum vessel, directly behind the first wall. The coils are mounted on supports installed on the inner surface of the vacuum vessel. A bridge consisting of two electrically insulated vertical conductors is installed to electrically connect the upper and lower rings. An analysis of plasma current disruption scenarios with rapid vertical plasma displacement showed that the passive stabilization coils must be completely isolated from the vacuum vessel. To facilitate the installation of coils in the vacuum vessel and their subsequent dismantling if necessary, the coils are segmented into several parts.

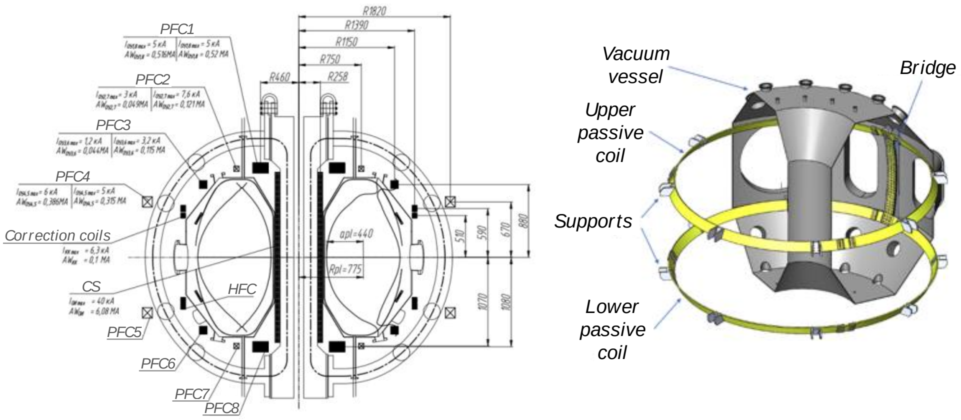


FIG. 2. Cross-section of the Globus-3 tokamak electromagnetic system (left). Potentially realized plasma column configurations with positive and negative triangularity are inscribed within the discharge chamber. The arrangement of passive stabilization coils within the vacuum vessel (right).

A preliminary baseline scenario for operation at a plasma current of 0.8 MA was developed for the tokamak electromagnetic system design presented above (see Fig. 3). The plasma current rise rate is 5 MA/s, with a plateau duration of 2-3 s. The central solenoid operates in a double swing mode. The stored volt-second capacitance is 2 Wb. The current range in the coils is approximately the same as in the currently operating Globus-M2 tokamak, allowing the Globus-3 tokamak electromagnetic system coils to be initially powered from existing power supplies. Subsequently, during the preliminary design stage, the feasibility of increasing the plasma current up to 2 MA and the toroidal magnetic field up to 2 T will be analyzed. As will be shown below, this will enable at least threefold increase in the plasma ion central temperature, to 15 keV, with a high average density.

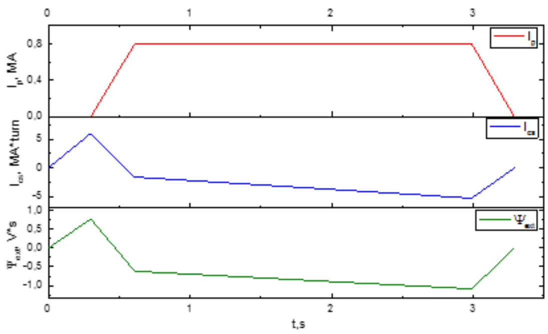


FIG. 3. Basic operating scenario of the Globus-3 electromagnetic system.

The preliminary design of the vacuum vessel is shown in Fig. 4. Austenitic stainless steel AISI 304 was selected as the primary structural material. The vacuum vessel's midplane contains ports for connecting four neutral beam injectors. Three of these injectors are designed for central plasma heating and have a 0.65 m impact parameter. One injector has a 0.95 m impact parameter and is intended for off-axis injection of high-energy atoms to generate plasma current at the plasma periphery, achieve off-center plasma heating, and generate an inverse fast ion distribution function. Opposite each injector, a trap is provided for atoms that pass through the plasma without ionization. The vessel also equipped with a large port for mounting an ion cyclotron resonance heating antenna or a lower-hybrid current drive grill. In addition to the equatorial ports, the tokamak has additional ports above and below the equator. These ports will house the gas injection systems, divertor diagnostics (thermal imager, pyrometer, divertor Thomson scattering diagnostics, and soft X-ray divertor diagnostics), interferometers with a vertical observation line, terminals for magnetic probes, Langmuir probes, and other in-vessel diagnostics, as well as spectrometric and soft X-ray diagnostics. The port locations are preliminary and may change during the development of the project design.

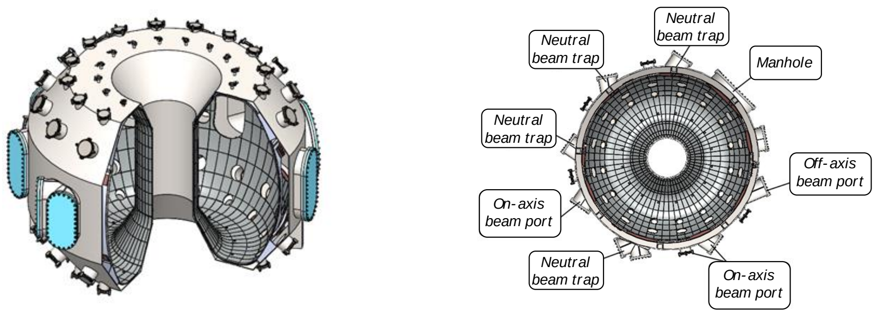


FIG. 4. Globus-3: sketch of the vacuum chamber (left) and top view (right).

Based on the experience of using in-vessel components in the Globus-M/-M2 facilities, graphite is also planned for the plasma-facing surfaces of the Globus-3 spherical tokamak. Graphite tiles are screwed to the mounting panels. The modular design of the in-vessel components and its universal mounting allow for the installation of items of various configurations, as well as their individual replacement if necessary.

3. AUXILIARY HEATING SYSTEMS

Tangential (on- and off-axis) neutral beam injection with a power of 10-12 MW is being considered as the primary method of auxiliary heating at the Globus-3 tokamak. Preliminary calculations have shown that in the regime with an average density of $n = 1\times10^{20}$ m^{-3}, the energy range of 50-60 keV is optimal. In this case, the value of direct power losses during injection will not exceed 10%. Corresponding injectors with a deuterium atomic beam power of ~3 MW are currently being developed at the Budker Institute (see Fig. 5). The ion source of the injector has a three-electrode ion-optical system (IOS) with slit structures. The initial beam size is 214 × 234 mm², and the designed ion beam current is 75 A. A distinctive feature of the electrodes in the Globus-3 injector is that they will be manufactured using selective laser melting technology. This technology allows for the production of electrodes with channels for active cooling and a spherical shape. Samples of flat bronze electrodes have already been produced using this technology. The spherical shape will ballistically focus the beam and make its size small enough, which is very important for medium-sized tokamaks with small ports. The plasma emitter in the ion source is formed by two 60 kW RF drivers operating at frequency of 4 MHz. The flows of plasma from the drivers are mixed in an expansion chamber connected to the IOS. A 2-meter-long neutralizer has been developed for the testbed version of the 100 keV beam injector. Since the optimal injection energy in Globus-3 is lower, this will allow the neutralizer length to be made shorter, thereby shortening the injection path and reducing the beam size at the tokamak port entrance, which is quite narrow in the toroidal direction.

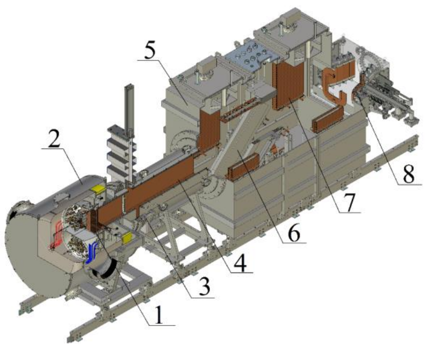
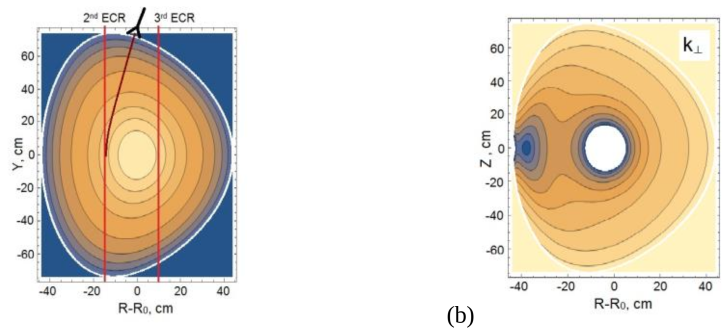


FIG. 5. Neutral beam injector: 1 – Ion Source, 2 –Magnetic Shield, 3 – Suspend Unit, 4 – Neutralizer, 5 – Vacuum Chamber, 6 – Separating System, 7 – Cryogenic Pump, 8 – Calorimeter.

The Globus-3 tokamak is expected to continue using the auxiliary heating and current drive systems previously performed on the Globus-M2 tokamak using ICR and LH waves. Specifically, a MultiJunction antenna (2.45

GHz) with an integrated power divider and phase-shifting inserts is being developed. This design is expected to be tested on the Globus-M2 and T-15MD tokamaks. Simultaneously, work is underway to create a lower-hybrid current drive system at 4.6 GHz, which will be tested on the FT-2 tokamak. Preparations are also underway for experiments on generating non-inductive current on the Globus-M2 tokamak using helicons. If successful, the system could be used on Globus-3.

Due to the increased magnetic field, it becomes possible to implement an electron cyclotron heating (ECRH) system in the tokamak. Microwave power from the gyrotron at a frequency of 110 GHz will be injected into the plasma through the upper port. In the expected high-density operating mode, with a central density of about $n_0 = 1\times10^{20}$ m^{-3} and higher, with a toroidal magnetic field at the center of $B_0 = 1.5$ T, the second and third harmonics of the EC resonance are accessible to extraordinary and ordinary waves. In both cases, microwaves propagate almost tangentially to the resonant surface, significantly increasing the absorption layer. With an expected central electron temperature of about 2-4 keV, ECR damping of the X2 mode should be single-pass, while that of the X3 and O2 modes should be multi-pass. Thus, ECR heating in the X2-mode is the most promising scenario. Figure 6 on the left shows the trajectory of the extraordinary wave calculated by ray tracing procedure in the poloidal cross section of Globus-3 ($n_0 = 1\times10^{20}$ m^{-3}, $T_e = 3$ keV, $B_0 = 1.5$ T). It is interrupted when the beam power becomes less than 1% of the launched power. Figure 6 on the right shows a contour plot of the perpendicular component of the wave number of an extraordinary wave in the poloidal cross section of Globus-3. The white region is evanescence area, which is due to the L-cutoff for the X-mode.



(a) (b)

FIG. 6. (a)The ray-traced trajectory of the extraordinary wave is shown by the brown curve in the poloidal cross-section of Globus-3 ($n_0 = 1\times10^{20}$ m^{-3}, $T_e = 3$ keV, $B_0 = 1.5$ T) and (b) the contour diagram of the perpendicular component of the wave number of an extraordinary wave in the poloidal cross-section of Globus-3. The white region is the evanescence region due to the L-cutoff for the X-mode.

4. EXPECTED PLASMA PARAMETERS

The plasma parameters were estimated for the baseline tokamak operation scenario for two injection power levels: at the initial stage of operation (~ 2 MW) and after commissioning of the new neutral beam injection complex (~ 12 MW). The hydrogen plasma temperature in the shots with hydrogen beam injection was calculated using the ASTRA code [16]. The ion transport coefficients were assumed to be neoclassical, while the electron transport coefficients were selected such that the energy confinement time coincided with twice (corrected for sphericity) τ_E calculated using the IPB98(y,2) scaling [7]. The effective charge was assumed to be 2, where the main impurity is carbon. The energy of the injected hydrogen atoms was 60 keV. An example of the ion and electron temperature profiles for the case of $B_{T0} = 1.5$ T, $I_p = 0.8$ MA at $\langle n_e \rangle_V = 10^{20}$ m^{-3} is shown in Fig. 7. Calculations have shown that when using the injectors available at the Globus-M2 tokamak, the ion and electron temperatures will be in the range of 1.5-2 keV. When the new neutral beam injection complex is put into operation, the ion and electron temperatures will be 5 and 2 keV respectively, i.e. approximately the same as in the hot ion mode at Globus-M2 [11], but at higher density. The uncertainty in the temperature profiles presented in the figure is due to the peaking factor of the density profile used in the calculation. Two cases with a peaking ($p = n_e(0)/\langle n_e \rangle_v$) of 1.1 and 1.6 were considered. A higher ion temperature corresponds to a flatter density profile.

Increasing B_{T0} to 1.8 T and I_P to 2.0 MA leads to a significant increase in the ion temperature (see Fig. 8). Two energy confinement time options were considered: a conservative one with τ_E using the IPB98(y,2) scaling and an optimistic one with a doubled confinement time of $2\times\tau_E$, which can be considered as a lower and upper estimation for the plasma parameters of the future facility. In both cases, the ion thermal diffusivity was assumed to be neoclassical. Calculations show that with the conservative approach, the central ion temperature reaches 10-20 keV. The share of non-inductive current will be 15-20%. The optimistic approach increases the ion temperature up to 20-40 keV and the share of non-inductive current to 25-35%. The equivalent neutron yield for D-T plasma in this mode will be at the level of 10^{16}-10^{18} s^{-1}. The figure also shows that in both cases considered, the central rotation velocity exceeds 100 km/s. It should be noted that similar rotation velocity values at the Globus-M2 tokamak resulted in the suppression of anomalous ion transport [12]. For the Globus-3 facility, the sufficiency of such rotation requires further study; however, it should be noted that, if necessary, the plasma rotation velocity can be increased by reducing the plasma density.

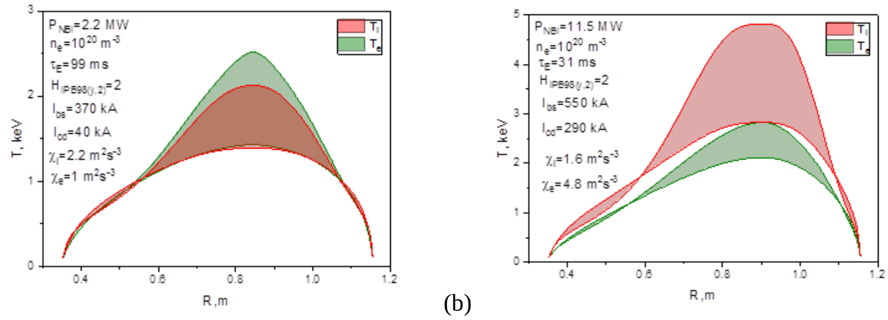


FIG. 7. Results of modeling the electron and ion temperatures in the Globus-3 tokamak for the conservative operating scenario B_{T0} = 1.5 T, I_P = 0.8 MA at an average plasma density $n_0 = 1\times10^{20}$ m^{-3}: (a) at an injection power of 2.2 MW, (b) at an injection power of 11.5 MW.

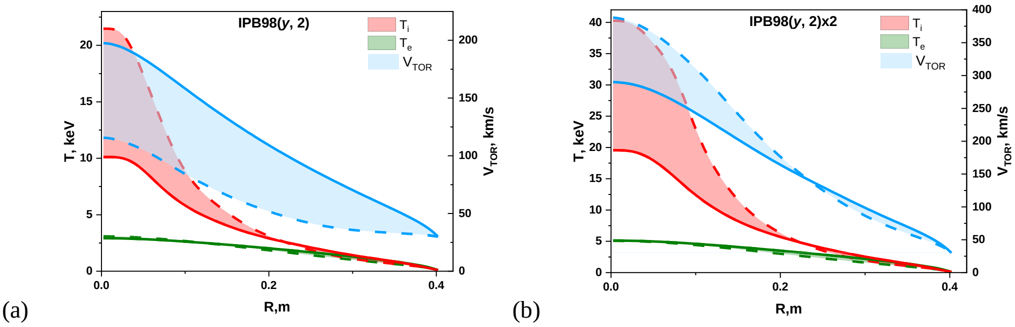


FIG. 8. Results of modeling the electron and ion temperatures and the velocity of toroidal plasma rotation at an average plasma density $n_0 = 1\times10^{20}$ m^{-3} for the scenario B_{T0} = 1.8 T, I_P = 2.0 MA at an injection power of 11.5 MW: (a) τ_E calculated using the IPB98(y,2) scaling (b) τ_E is twice as high as that calculated using the IPB98(y,2) scaling. The boundaries on the graphs correspond to the peaked density profile p=1.1 (dashed line) and p=1.6 (solid line).

An analysis of the dependence of plasma parameters on injection power (see Fig. 9) showed that, in the case of neoclassical ion behavior, the electron temperature T_e remains virtually unchanged with increasing power, the bootstrap current I_{bs} is saturated at 6 MW, and the ion temperature T_i and beam-driven current I_{CD} increase almost linearly. Off-axis injection has no significant effect on ion temperature and is useful for suppressing instabilities and generating current.

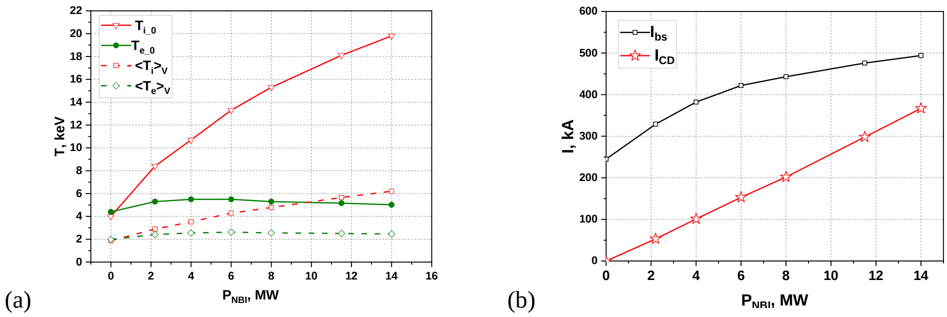


FIG. 9. Dependence of plasma parameters in the Globus-3 tokamak (I_P = 2,0 MA, B_0 = 1.8 T, $n_0 = 1\times10^{20}$ m^{-3}, peaking factor p=1.6, Neoclassical ion transport, IPB98y,2 scaling, H-factor = 2) on the injection power: (a) temperature of ions and electrons, (b) non-inductive plasma current.

The Globus-3 tokamak provides the opportunity to study the behavior of thermonuclear alpha particles under non-nuclear experimental conditions. Calculations show that alpha particles in the MeV energy range will be confined. Since work with tritium is not planned at Globus-3, alpha particles may be produced as a result of the p + ^{11}B -> 3^{4}He reaction. This reaction produces alpha particles with a broad spectrum from approximately 1 to

5.7 MeV. The spatial distribution of the fraction of confined alpha particles for two main peaks with energies of 2.46 MeV and 3.76 MeV at B_{T0} = 1.8 T, I_p = 2 MA and B_{T0} = 1.5 T, I_p = 0.8 MA is shown in Fig. 10. The calculations assumed an isotropic distribution of the alpha particle source. As can be seen from the figure, at B_{T0} = 1.8 T, I_p = 2 MA, more than 90% of the generated alpha particles are confined in the central region of the plasma, while even with reduced parameters (B_{T0} = 1.5 T, I_p = 0.8 MA), a confined fraction of alpha particles will exist near the magnetic axis.

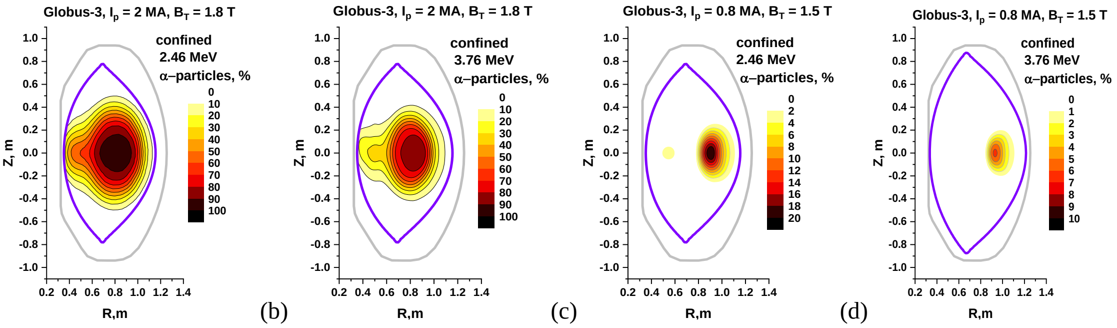


FIG. 10. Fraction of confined α-particles produced in different regions of the Globus-3 tokamak plasma. Confinement at B_{T0} = 1.8 T, I_p = 2 MA of α-particles with energies of 2.46 MeV (a) and 3.76 MeV (b). Confinement at B_{T0} = 1.5 T, I_p = 0.8 MA of α-particles with energies of 2.46 MeV (c) and 3.76 MeV (d). The gray line is the tokamak chamber, the purple line is the last closed magnetic surface.

A significant increase in the discharge parameters in the Globus-3 tokamak may lead to an increase in the load on the first wall. Preliminary modeling of the edge plasma parameters for a conservative scenario for the Globus-3 tokamak was performed using the SOLPS-ITER version 3.2.0 code on an unstructured mesh covering the entire volume of the vacuum vessel [18]. The modeling was carried out without taking into account drifts using the EIRENE Monte Carlo code [19] for modeling the behavior of atoms and molecules. The discharge power, and therefore the heat flux entering the computational domain across the inner mesh boundary, was set to 1.5 MW. Deuterium was used as the working gas in the calculations, and carbon sputtered from the surface of the first wall was used as an impurity. The anomalous transport coefficients used as input parameters for the simulation were taken from the Globus-M2 tokamak shot #44644, where they provided good match between the simulated and experimental profiles of the electron density, electron and ion temperatures on the outer midplane. The working gas puffing rate was 3×10^{20} atoms per second. Since the simulation considered a short-duration discharge scenario, largely similar to Globus-M2 discharges, the particle sink was implemented as deuterium absorption by the graphite wall with a reflectivity of 99.3% for deuterium incident on the material surface. The simulation results are presented in Fig. 11. For the given discharge parameters, the thermal load on the divertor plates did not exceed 1.5 MW/m^2.

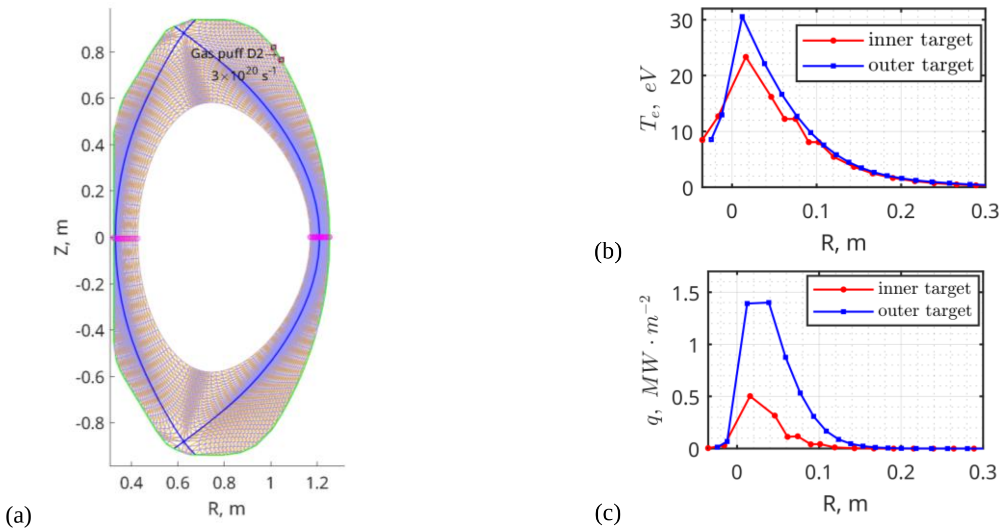


FIG. 11. Simulation of the thermal load on the first wall in the Globus-3 tokamak: (a) SOLPS-ITER (blue) and EIRENE (orange) meshes, pink constrictions indicate the position of the outer and inner midplanes; (b) electron temperature and (c) heat flux density at the lower divertor targets. Zero on the abscissa in the figures on the right corresponds to the position of the separatrix.

5. CONCLUSION AND FUTURE PLANS

Research experience with the Globus-M/-M2 tokamak and other spherical tokamaks has provided the engineering and physical basis for the creation of next-generation facilities in the near future. Analysis of plasma heating and confinement clearly demonstrates that the size of the facility is no less important than the magnitude of the toroidal magnetic field. The presented Globus-3 tokamak design envisions a simultaneous increase in both B_T and R. It will operate in a previously unexplored parameter range, which, in our opinion, will enable the highest plasma confinement performance. Modeling results demonstrate, in particular, the feasibility of producing plasma with near-thermonuclear temperatures and very low collisionality, allowing the Globus-3 tokamak to be considered as a hydrogen prototype for a fusion neutron source. Furthermore, the facility will remain relatively compact, ensuring its relatively low cost and, importantly, short construction and experimental preparation times.

The above parameters of the Globus-3 tokamak for the first phase of experiments are quite conservative. This, if funding is allocated, allows for its rapid design (the technical design is expected to be completed in 2027) and manufacture within a relatively short period of 5-6 years, followed by rapid beginning of experiments. During the next phase of experiments, it is planned to retrofit the tokamak with new, powerful systems for auxiliary heating and non-inductive current drive, significantly increasing the specific heating power. The cost of such systems is quite high and comparable to the cost of the tokamak itself. However, the situation is somewhat mitigated by the closeness of the engineering parameters of these systems to those of similar systems in the T-15MD tokmak, which is currently being commissioned. This will accelerate their development and reduce costs. Further consideration is planned for conceptual designs of a facility with a magnetic field exceeding 2.5 T and a plasma current of 2 MA, with the possible use of superconductors in the electromagnetic system design. The possibility of such modernization will be provided in the original design of the Globus-3 tokamak.

ACKNOWLEDGEMENTS

The work uses the results of research conducted at the Unique Scientific Facility "Spherical tokamak Globus-M", which is incorporated in the Federal Joint Research Center "Material science and characterization in advanced technology" of the Ioffe Institute.

The design of the electromagnetic system and vacuum vessel was carried out with financial support of the Russian Science Foundation (Agreement № 21-79-20133-P dated 28.05.2025). Fast ion study was carried out with financial support of the Russian Science Foundation (Agreement № 21-72-20007-P dated 28.05.2025).

REFERENCES

[1] GUSEV V.K., et al., Tech. Phys., **44** (1999) 9, 1054-1057.
[2] PENG Y-K.M, STRICKLER D.J., Nucl. Fusion, **26** (1986) 769.
[3] BAKHAREV N.N., et al., Nucl. Fusion, **55** (2015) #043023.
[4] VALOVIC M., et al., Nucl. Fusion, **49** (2009) #075016.
[5] KAY S.M., et al., Nucl. Fusion, **46** (2006) 848-857.
[6] BAKHAREV N.N., et al., Nucl. Fusion, **58** (2018) # 126029.
[7] ITER Physics Basis, Nucl. Fusion **39 (**1999) 2175-2249.
[8] DOYLE E.J., et al., Nucl. Fusion**, 47** (2007) S18-S127.
[9] MINAEV V.B., et al., Nucl. Fusion, **57** (2017) 6, #066047.
[10] GUSEV V.K., et al., Nucl. Fusion, **53** (2013) 9, #093013.
[11] KURSKIEV G.S., et al., Nucl. Fusion, **62** (2022) #016011.
[12] KURSKIEV G.S., et al., Nucl. Fusion, **62** (2022) #104002.
[13] KURSKIEV G.S., et al., Phys. Plasmas, **31** (2024) #062511.
[14] MINEEV A.B., et al., Physics of Atomic Nuclei, **85** (2022) S17–S33.
[15] MINAEV V.B., et al., Plasma Physics Reports, **49** (2023) 1578–1587.
[16] PEREVERZEV G.V., YUSHMANOV P.N., IPP Report, 2002.
[17] KURSKIEV G.S., et al., Nucl. Fusion, **61** (2021) #064001
[18] DEKEYSER W., et al., Nucl. Mater. Energy, **27** (2021) 100999
[19] https://eirene.de/Documentation/eirene.pdf