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

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

[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

In 1999, the first Russian spherical tokamak Globus-M [1] ($R \approx 0.36$ m, $A \approx 1.5$) with 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 ($\rho_i{}^* = \rho_i / a$) was put into operation at the Ioffe Institute. The neutral beam plasma heating system was introduced after the facility was taken in operation and provided a high specific power of auxiliary heating. However, the heating efficiency was negatively affected by poor energy confinement and high level of high-energy ion losses due to the low magnetic field [2]. As was discovered at the MAST [3] and NSTX [4] facilities and later confirmed at the Globus-M tokamak [5], the energy confinement time in a spherical tokamak, unlike a conventional one [6,7], has a strong dependence on the magnetic field. The electromagnetic system of the next tokamak Globus-M2 [8,9], launched in 2018, was completely redesigned, 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. As a result, a significant improvement in energy confinement [10] and the obtaining of a "hot ion" mode [11], in which the ion temperature exceeded 4 keV, were achieved. The analysis showed that ion heat transport in this mode is well described by the neoclassical theory [12]. This report presents the design of the Globus-3 tokamak [13,14], 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. Distinctive features of the facility include the increased toroidal magnetic field, the larger torus major radius and the duration of the plasma discharge, which exceeds the characteristic times of stationary plasma parameter profiles formation. All auxiliary heating and current drive systems (NBI, ICRH, LHCD) previously used in the Globus-M/-M2 tokamaks must be explored. In addition, increasing the magnetic field allows the implementation of an electron-cyclotron heating system. The modeling results show the possibility of obtaining plasma with a near-fusion temperature and very low collisionality, which allows considering the Globus-3 tokamak as a hydrogen prototype of a compact fusion neutron source.

At the pre-conceptual design stage, three options for the electromagnetic system were considered: with warm copper coils, with copper coils pre-cooled to the temperature of liquid nitrogen, and with coils made of high-temperature superconductors. The last two were rejected due to the lack of technology readiness and problems with accommodating the facility in the existing machine hall. Limitations associated with the available electrical network power supply capacity of 125 MVA were also taken into account. The provided analysis of the parameters and the development of the facility 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 $\Delta_{tplateau}$ = 2-3 s; pause between pulses ≤ 30 minutes; toroidal field coil overheating per pulse ≤ 40°C; gap between plasma and vacuum vessel on the high field side 25 mm; number of toroidal coils $N_{TF} = 16$; value of the toroidal field ripple on the low field side $\Delta_{ripple} \leq 0.4\%$. We also consider an ultimate scenario with increased values of magnetic field $B_{T0} = 1.8$ T and plasma current $I_P = 2.0$ MA. In general, the layout of the components of the electromagnetic system and the vacuum vessel of the Globus-3 tokamak (see fig. 1) conceptually repeats that of the Globus-M2 tokamak. Also it is proposed to implement the passive system to stabilize the vertical plasma position by means of two coils located inside the vacuum vessel in the upper and lower zones directly behind the invessel components. The overall dimensions of the Globus-3 facility are 3.8 × 3.8 m (transverse size × height), which makes it possible to place the tokamak in an existing machine hall.

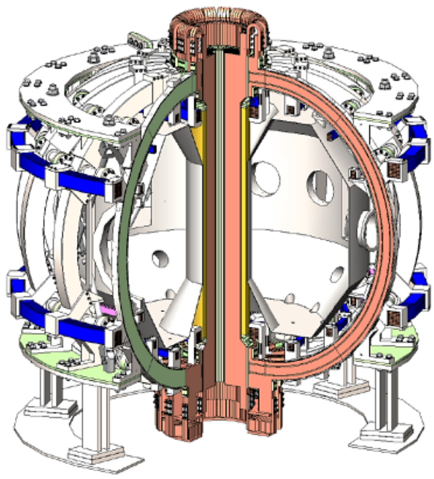

Figure 1. Preliminary design of the electromagnetic system and the vacuum vessel of the Globus-3 tokamak

For the above-presented design of the electromagnetic system of the tokamak, a preliminary operating scenario at a plasma current of 0.8 MA was developed. The range of currents in the coils is approximately the same as in the currently operating Globus-M2 tokamak, due to which the electromagnetic system of Globus-3 can be fed by existing power supplies at the first stage. The central solenoid operates in the double-swing regime. The stored volt-second capacity will be 2 Wb.

Tangential (on-axis and off-axis) neutral beam injection with a power of 10-12 MW is considered as the main method of auxiliary heating. The Budker Institute is currently developing appropriate injectors. The plasma parameters are estimated for conservative and ultimate scenarios. For the case of B_{T0} = 1.5 T, I_P = 0.8 MA at an average density of $\langle n_e \rangle \sim 10^{20}$ m^{-3}, the temperatures of ions and electrons will be about 5 and 2 keV, respectively, i.e. approximately as in the "hot ion" mode at Globus-M2 [11]. Increasing B_{T0} to 1.8 T and I_P to 2.0 MA will lead to a significant growth in the ion and electron temperatures to 10-30 and 3-10 keV, respectively (depending on the scaling used: IPB 98(y,2), or Globus-2021). This mode corresponds to an equivalent neutron yield in D-T plasma at the level of 10^{16}-10^{18} s^{-1}. Also, under such conditions it will become possible to study the behavior of thermonuclear alpha particles. A significant increase in plasma parameters in the Globus-3 tokamak will obviously lead to power load rise on the first wall. The SOL plasma simulation was performed using the SOLPS-ITER code for different divertor configurations. The impurity gas puffing for protecting the divertor plates and the first wall were investigated.

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