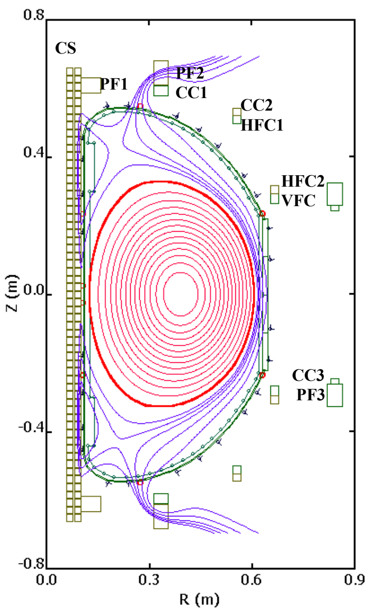
INVESTIGATION OF EQUILIBRIUM IN GLOBUS-M OHMIC PLASMAS

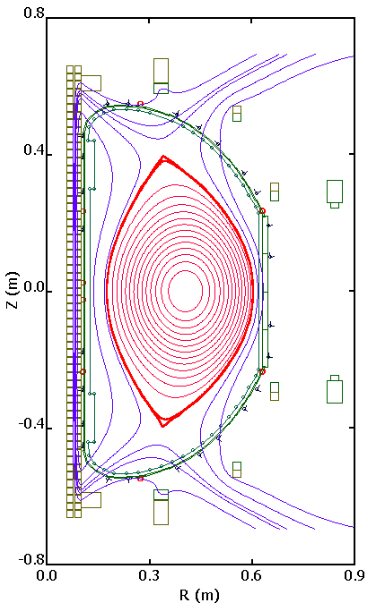
N.V.Sakharov[1], S.E.Bender[2], V.E.Golant[1], V.K.Gusev[1], A.V.Dech[1], Yu.A.Kostsov[2], E.A.Kuznetsov[3], R.G.Levin[1], V.B.Minaev[1], A.B.Mineev[2], O.A.Minyaev[1], A.N.Novokhatskii[1], Yu.V.Petrov[1], E.N.Rumyantsev[2], V.N.Scherbitsky[3], V.I.Vasil'ev[2], V.A.Yagnov[3].

[1] - Ioffe Physico-Technical Institute, 26 Politekhnicheskaya, 194021, St.Petersburg, Russia
[2] - D.V.Efremov Institute, 196641, Metallostroy, St.Petersburg, Russia
[3] - TRINITI, 142092, Moscow Region, Troitsk, Russia

Spherical tokamak Globus-M is designed to investigate plasmas with major radius R=0.36 m, minor radius a=0.24 m, aspect ratio A=R/a=1.5, plasma current up to 0.5 MA in toroidal magnetic field up to 0.6 T on axis, elongation up to 2.2, triangularity up to 0.4. The description of Globus-M can be found in [1]. The full-scale experiments with Ohmic plasmas were started in the summer of 2000, and then were continued in the winter of 2001 [2]. The plasma current up to 0.25 MA was achieved in double X-point magnetic configuration. The experimental range of vertical elongation is 1.1-1.9, and triangularity 0.1-0.4.



(a)



(b)

Fig.1. EFIT reconstructions of poloidal flux contours for various ratios of current in central solenoid and in CC.
a) inner wall limited, k = 1.4
b) double X-point, k = 1.9

Fig.1a shows the position of the vacuum vessel, the central solenoid (CS) and the PF coils. Plasma in Globus-M is confined within a thin-wall stainless steel vacuum vessel with characteristic wall thickness of 2-3 mm. In the first plasma experiments the number of commissioned power supplies was insufficient for energizing of all the PF coils. The central solenoid operated in a preprogrammed single swing regime, and the magnetic flux ramp-up phase was used for plasma breakdown and maintenance. The coils CC1-CC3 used for compensation of the central solenoid stray magnetic field were connected in series. The PF3 provided the main part of vertical magnetic field required for plasma equilibrium along the major radius. Coils PF1 and PF2, designed for plasma shaping were not utilized in the experiments. Under these conditions the central solenoid stray field was used for variations of plasma shape. For this purpose, current in compensation coils CC1-CC3 was changed during plasma discharges, while the preprogrammed current in the solenoid remained the same. Scenarios of Globus-M shots are shown in Fig.2. The waveforms of current in the central solenoid are the same for both shots 2390 and 2398. The loop voltage is measured at the radius R = 0.11 m in the mid-plane. The integrated value of V_l gives the central solenoid magnetic flux, which did not exceed 0.1 Wb in the last experimental campaign.

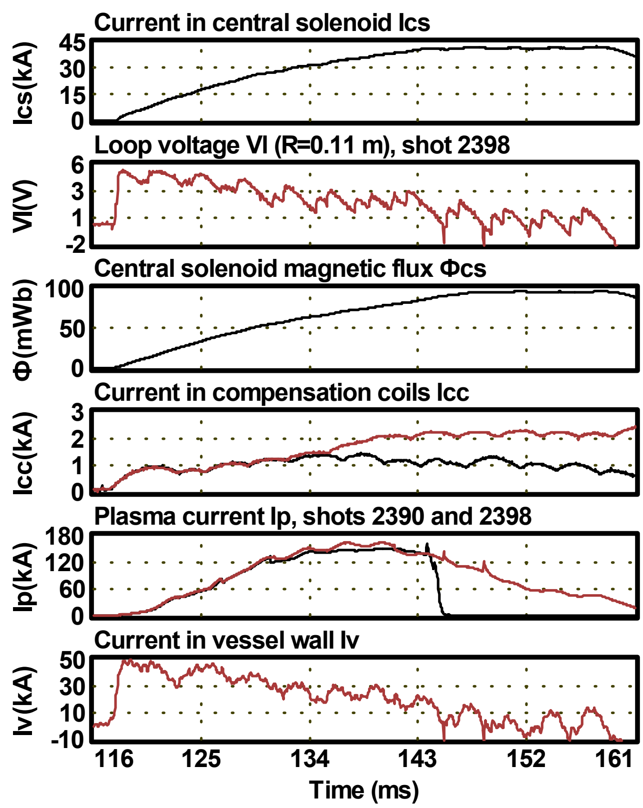


Fig.2. Time evolution of current in central solenoid, loop voltage, solenoid magnetic flux, currents in CC1-CC3, plasma currents and current in vessel walls in the shots with various plasma vertical elongation.

In the shot 2390, the current in compensation coils was reduced, which led to the increase of plasma vertical elongation and even to termination of the plasma discharge after a few internal reconnection events (IRE) described in previous ST experiments [3]. Plasma magnetic configurations for different values of currents in CC1-CC3 are shown in Fig.1a and Fig.1b. The equilibrium configurations are the result of EFIT code [4] analysis between shots. The input data comprised signals of about 30 magnetic probes measuring normal and tangential components of the poloidal magnetic field, plasma current, coil currents and toroidal current in the vacuum vessel. The plasma current was measured with the Rogowsky coil positioned inside the vacuum vessel. The second Rogowsky coil was placed outside the vessel and recorded the total toroidal current. The difference between two measured currents gives the current induced in the vessel walls (see Fig.2). The spatial distribution of current in the vessel was simulated with the help of an eddy current model. The model was developed on the basis of the detailed measurements of the poloidal field produced by each pair of PF coils and the central solenoid inside the vessel in the absence of plasma. During the discharge the induced current in the vessel wall varies between 30-50 kA in the phase of plasma current ramp-up and 5-20 kA in the phase of plateau and ramp-down phase. The difference between the values of the experimentally measured plasma current and EFIT reconstructed one is within 2-4%.

Plasma in Globus-M is tightly inserted into the vessel. The gap between the plasma outer boundary and the first wall is about 2-4 cm which requires a careful control of plasma position. The vessel L/R time is near 1-2 ms. Plasma R and Z positions are sustained by the feedback control system based on analog amplifiers and thyristor choppers with a frequency response up to 3 kHz. The choppers feed low inductance VFC for R-position control and two pairs connected in series HFC1 and HFC2 for Z-position control (see Fig.1).

The algorithm of R-position control utilizes average flux measured by two saddle loops located on the vessel inner surface below and above the mid-plane, and two magnetic probes. The saddle loops are extended along the toroidal axis within the sector of 90^0. The characteristic distance between the loops and the plasma column in the mid-plane is 2 cm for plasma inner boundary and 2-4 cm for outer plasma boundary depending on plasma size. The inner and outer probes are located in the mid-plane at the same distance to plasma as the saddle loops. The linear equations combining signals of saddle loops and probes, or signal of saddle loops and measured plasma current are used to calculate the shift of plasma geometrical center $\Delta R = R - R_0$ relatively the center of the flux loop which practically coincides with the radius of vessel toroidal axis $R_0 = 0.37$ m.

The described method was analyzed by means of equilibrium simulation code. In the simulations, the magnetic flux through the saddle loops and signals of the probes were created by plasmas with various radial position, current, shape etc. These code - simulated data were processed then by the simple equations for ΔR. The calculated in such a way values of ΔR were in a good agreement with initial plasma displacements in code simulated configurations and appeared to be low sensitive to the plasma shape, beta factor and internal inductance at least within the range of $\Delta R = \pm 10$ cm.

The difference of signals measured by two flux loops positioned on the vessel top and bottom domes at the radius R = 0.28 m is used for plasma Z-position control.

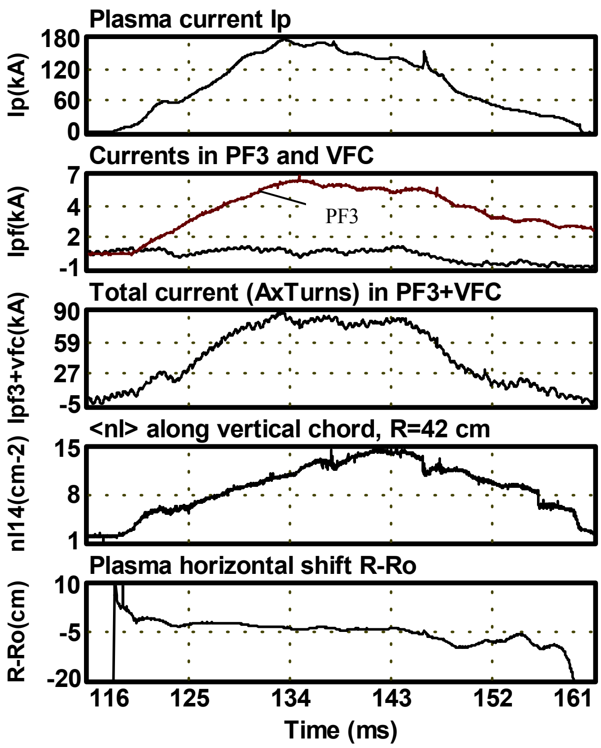


Fig.3. Time evolution of parameters, illustrating operation of R-position control system in a discharge with internal reconnection event.

Fig.3 illustrates the operation of the system for plasma radial position feedback control. For the value of the plasma current ~ 0.2 MA VFC can provide up to 30% of the vertical magnetic field required for the plasma equilibrium along the major radius. These coils are used for the control of the plasma radial shift according to the preprogrammed waveform. Coils PF3 are energized by “slow” six-phase thyristor rectifier and produce the main part of the vertical field. The principle of PF3 operation is to sustain current in VFC near zero level. During the shot the control system sustains the plasma column near the preprogrammed position within the accuracy about ± 0.5 cm. One can see the IRE spontaneously occurred in the second half of the discharge. The IRE causes radial displacement of ~ 3 cm towards the vessel inner cylinder. The control system responds with a frequency of thyristors commutation, which is about 1.5 kHz in the given case. However, the plasma is returned to the initial position after a few milliseconds delay caused by a limited productivity of the presently utilized power supply. The time evolution of the total current in PF3 and VFC in Fig.3 basically follows the waveform of the plasma current and also responds to the increase of the plasma density measured by a 1mm interferometer along the vertical chord positioned at the radius R = 42 cm.

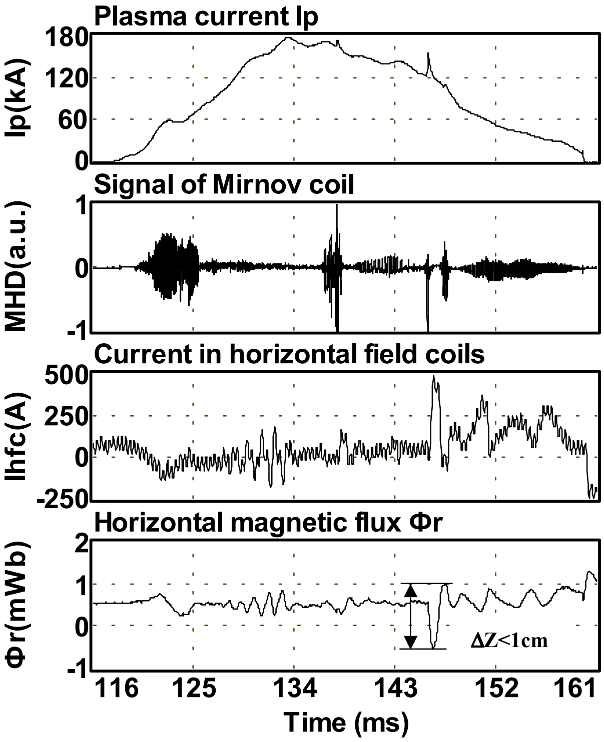


Fig.4. Time evolution of parameters, illustrating operation of Z-position control system in a discharge with internal reconnection event.

Fig.4 illustrates the operation of the plasma vertical position feedback control system. The thyristor chopper operating at a frequency ~ 2.5 kHz controls current in horizontal field coils in order to sustain the measured radial magnetic flux Φ_r near zero level. Ohmically heated plasmas in Globus-M with low values of elongation k ~ 1.2-1.5 are mostly vertically stable even in the absence of operating control system. In Fig.4, the control system responds to the bursts of MHD activities and IREs. During the IRE the estimated vertical shift does not exceed 1 cm. At higher values of elongation IREs could lead to termination of the plasma discharge accompanied by fast plasma movement towards the vessel inner limiter and by vertical displacement. The estimated values of plasma displacement towards the vessel upper dome could achieve 6-10 cm during IRE.

Achieved parameters and plans

The achieved range of plasma parameters comprises the plasma current up to 0.25 MA, the toroidal field in the range of 0.07-0.38 T, the plasma pulse length up to 60 ms, the vertical elongation of 1.1-1.9, the triangularity of 0.1-0.4, the safety factor $q_{95} \geq 2.1$, the minimum q_{cyl} ~0.9.

The nearest plans include an increase of the solenoid magnetic flux up to 0.25 Wb and the toroidal field up to 0.5-0.6 T. PF1 and PF2 will be utilized for plasma shaping. Digital control of plasma position is planned at the end of 2001.

Acknowledgements
This work was supported by ISTC, RF Ministry of Industry, Science and Technology, IAEA grant and RFBR grants 00-02-16934 and 01-02-17882.

References

[1] Gusev V.K., *et al,* Technical Physics, v.44, No.9, 1054, 1999
[2] V.K.Gusev, *et al,* Nuclear Fusion **41** (2001), No.7
[3] A.Sykes, *et al.,* Nucl. Fusion **32** (1992) 694.
[4] L.L.Lao, *et al.,* Nucl. Fusion **25** (1985) 1611.