Heating and Confinement Study of Globus-M Low Aspect Ratio Plasma

N.V. Sakharov 1), V.V. Dyachenko 1), B.B. Ayushin 1), A.V. Bogomolov 2), F.V. Chernyshev 1), V.K. Gusev 1), S.A. Khitrov 1), N.A. Khromov 1), G.S. Kurskiev 1), V.B. Minaev 1), M.I. Mironov 1), M.I. Patrov 1), Yu.V. Petrov 1), V.A. Rozhansky 2), I.Yu. Senichenkov 2), O.N. Shcherbinin 1), S.Yu. Tolstyakov 1), E.G. Zhilin 3)

1) Ioffe Physical-Technical Institute of the Russian Academy of Sciences, 194021, St.Petersburg, Russia.
2) St. Petersburg State Polytechnical University, 195251, St. Petersburg, Russia.
3) Ioffe Fusion Technology Ltd., 194021, St. Petersburg, Russia.

e-mail contact of main author: Nikolay.Sakharov@mail.ioffe.ru

Abstract. Results on plasma heating in experiments with neutral beam injection (NBI) and ion cyclotron resonance heating (ICRH) at the fundamental harmonic of hydrogen minority in deuterium plasma in the Globus-M spherical tokamak are presented. Numerical simulation of fast ion trajectories was performed in the plasma configurations reconstructed from magnetic measurements with the help of the EFIT code. Energy spectra of slowing down fast ions were measured with tangentially and perpendicularly directed neutral particle analysers in a wide range of energies. Reliable H-mode transition was achieved after the change of the toroidal magnetic field direction. The ion drift now is directed towards the lower X-point, which exists inside the vessel even in a limiter configuration. The most stable H mode transition was achieved in NBI experiments. Also L-H transition was observed in ohmic heating (OH) regime and at ICRH in limiter and divertor magnetic configurations. By means of multi-channel multi-pulse Thomson scattering (TS) diagnostic we observed a formation of a steep density gradient after the L-H transition. Flat density distribution in the plasma bulk was sustained during all the H-mode period. Modeling of transport processes in the L and H-modes was performed with the help of the ASTRA code. The calculated energy confinement times are consistent with the ITER scalings. It is shown that in the ion heat conductivity remains neoclassical the H-mode and no barrier is formed for the ion temperature profile. The particle diffusion coefficient is reduced sufficiently inside the transport barrier region.

1. Introduction

In this presentation we describe the results of plasma confinement and heating study in the spherical tokamak Globus-M. The presentation topic is focused on the plasma L-H transition, the confinement of fast ions during the plasma auxiliary heating, the simulation of the transport processes in the plasma bulk and the behavior of fast ions. The basic method of the plasma auxiliary heating is the neutral beam injection of a deuterium or a hydrogen beam up to 1 MW power in the 18÷28keV energy range [1]. The beam has an impact radius of 0.3 m. We used a tangential co-injection relatively the plasma current direction. The second auxiliary heating method is the ion cyclotron heating of 7÷9 MHz frequency corresponding to the fundamental harmonic of a hydrogen minority in deuterium plasma [2]. The ICRH power is about 0.2 MW.

The implementation of these auxiliary heating methods in the spherical tokamak Globus-M has some characteristic features. The first one is a high specific heating power (the power per plasma volume unit). A typical plasma volume is about 0.4÷0.5 m^3 depending on the magnetic configuration shape. Taking into account the OH power the total specific power achieves 2÷3 MW/m^3. Under these conditions a careful vacuum vessel conditioning is required. At present up to 90% of the plasma faced surface is plated with graphite tiles of the RGT grate, i.e. the recrystallized graphite with a low porosity and low spattering coefficients. The second feature is associated with a confinement of fast ion in small size plasma of the low toroidal magnetic field tokamak. The plasma parameters are: the major radius R~0.36÷0.33 m, the minor radius a~0.21÷0.24 m, the aspect ratio A=R/a~1.5÷1.7, the plasma vertical elongation κ~1.5÷2. At present the total current in the toroidal magnetic field coils I_{TF}≈0.7 MA. For this current the toroidal magnetic field drops along the major radius from 1.1 T to 0.23 T inside the plasma volume. Under these conditions the characteristic scale of fast ion orbits initiated in the period of NBI and ICRH becomes comparable with the plasma size. The energy spectra of fast non-maxwellian ions were studied by means of charge-exchange (CX) diagnostic complex. The diagnostic consists of two neutral particle analyzers (NPA) of ACORD type [3]. The NPA positions as well as the position of NBI, ICRH feeder and the TS diagnostic arrangement are shown in Fig.1. The ACORD-12 analyzer is installed perpendicularly to the toroidal axis, the analyzer ACORD-M – tangentially. The application of the new ACORD-M device allows obtaining the information about passing ions, including fast ions appeared in the NBI period [4].

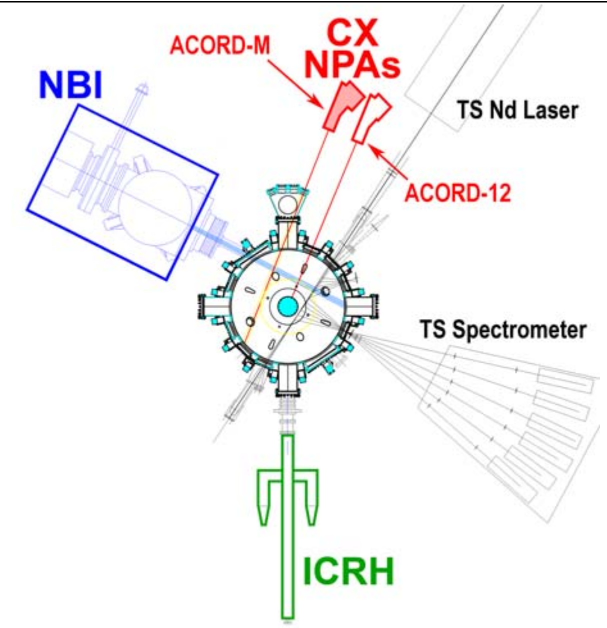


FIG. 1. Apparatus arrangement at the Globus-M tokamak.

Under the conditions when the neural beam vertical size and the characteristic scale of fast ion orbits are comparable with the plasma size, the transport simulation by means of numerical codes is facilitated. For simulation of the plasma heating and confinement in the Globus-M we use the ASTRA transport code [5] with a specially modified model of the neutral beam injection.

The H-mode transition was achieved after the toroidal magnetic field direction change [6]. At present the toroidal magnetic field is parallel to the plasma current. The ion gradB drift is directed toward the lower X-point. The L-H transition was observed in ohmically heated and auxiliary heated plasmas in limiter and X-point magnetic configurations. The change of the toroidal magnetic field direction itself improved the plasma performance even in the L-mode regime. In the experiments we observed approximately two-fold decrease of radiation losses, 20% decrease of the plasma internal inductance, 20÷30% reduction of the plasma voltage accompanied by some increase of the plasma energy in OH regime. This result means also an improvement of the plasma energy confinement in L-mode. The reasons of the described effects are not quite clear. A possible explanation can be connected with the influence of magnetic error fields. Particular, many experiments demonstrate a tendency to the plasma stability improvement in a lower X-point configuration, or when the plasma center is shifted a few centimeters below the midplane. The most stable L-H transition was achieved in the NBI heated deuterium plasma.

The results of the experiments and simulation are described below.

2. Study of confinement of fast ions

2.1 Ion cyclotron heating

ICRH experiments are performed in deuterium plasma with hydrogen minority. The resonance frequency corresponds to the hydrogen fundamental harmonic. The absorption of

ICR emission in plasma is accompanied by generation of hydrogen supra thermal ions. CX energy spectra of hydrogen and deuterium measured by ACORD-12 analyzer (see Fig.1) are shown in Fig.2. The spectra analysis reveals that the effective temperature of the tail observed in the hydrogen energy spectrum is restricted, and the maximum value is T_{tail}~1.2 keV. Due to a low signal the measurements are performed at the energies not higher than 5 keV. However, the analysis of CX fluxes decay after the ICRH pulse termination gives the highest possible energy of accelerated H-ions confined in plasma as E_{max}~ 15 keV [7]. In order to confirm the estimates the fast ion orbits were simulated. The example of the banana orbit of 10 keV hydrogen ion accelerated perpendicular to the toroidal axis is shown in Fig.3. In the case of Fig.3 the plasma current is I_P=0.2 MA and the toroidal magnetic field is B_T=0.4 T. For the orbit simulation we used EFIT reconstruction of magnetic equilibrium [8]. Red line indicates the plasma outmost closed magnetic surface. The red-blue line shows the cyclotron resonance zone for the hydrogen fundamental harmonic ω_H.

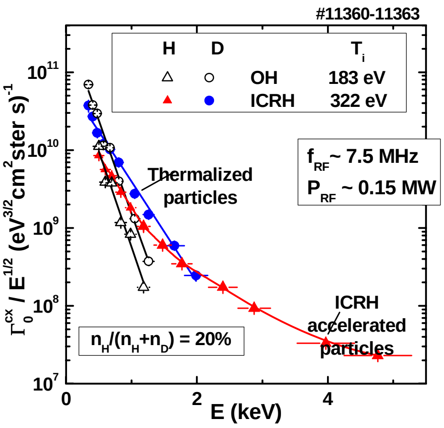


FIG. 2. OH and ICRH CX energy spectra.

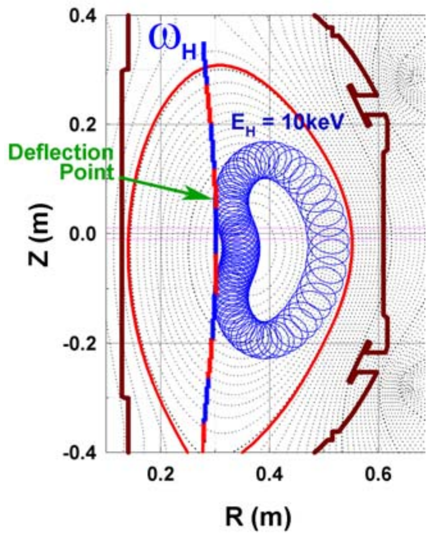


FIG. 3. Banana trajectory of 10 keV hydrogen ion.

2.2 NBI heating

In comparison with large tokamaks operating in a strong toroidal magnetic field the problem of the confinement of fast ions is critical for the NBI heating experiments in Globus-M. Estimates predict noticeable first orbit losses of the beam ions and also shine through losses at low plasma density in a low toroidal magnetic field and in a small size plasma. For these reasons study of the confinement of fast ions becomes especially important for the experimental conditions optimization.

The ion temperature of deuterium and hydrogen in the plasma bulk is measured with CX analyzer ACORD-12. The second analyzer ACORD-M (see Fig.1) allows measuring the energy spectra of passing ions in a wide energy range up to the injected beam energy. A numerical modeling is used for the experimentally measured energy spectra analysis and interpretation. Similar to ICRH experiments a simulation of fast ion orbits was performed and a spatial zone of the confinement of fast ions was outlined for some range of plasma magnetic configurations and different values of the neutral beam impact radius [7].

A deuterium or hydrogen beam was co-injected into deuterium plasma. First results of the slowing down energy spectra measurements are shown in Fig.4. The spectra in Fig.4 are measured at 10 ms after the beginning of NBI. For better presentation the spectrum for the

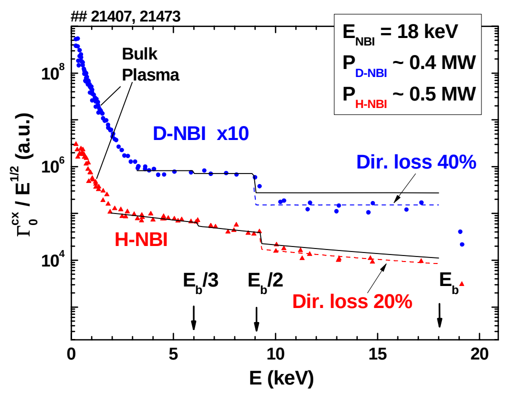


FIG. 4 Slowing down energy spectra measured during H-NBI and D-NBI.

case of the deuterium beam injection is multiplied by a factor of 10. The solid line corresponds to the theoretical prediction of slowing down due to coulomb collisions [9]. The dashed line is the theoretical prediction taking into account direct losses of the main beam component. In total the shapes of both spectra well correspond to the slowing down due to coulomb collisions and the results of beam particle trajectory simulations, which predict the direct loss level of 20 % for H-beam and 40 % for D-beam. Note that the CX spectra in Fig.4 are measured at relatively low density in order to reduce dumping of CX fluxes from the plasma center. The heating of the plasma bulk appeared to be approximately the same for deuterium and hydrogen beam injection with an equal NBI power. The bulk plasma heating is described in the next section.

3. Study of low aspect ratio plasma in improved confinement regime

3.1. Ohmic H-mode

Fig.5 shows a time evolution of plasma parameters in the process of L-H transition in ohmically heated plasma. The transition takes place in the initial phase of the plasma current plateau. It is accompanied by a spontaneous decrease in D_α emission and a rise of the plasma average density. The electron density in Fig.5 is measured by 0.8 mm interferometer along the vertical chord positioned at the radius R=0.42m. In the case of Fig.5 the L-H transition occurs in a lower X-point plasma magnetic configuration. The magnetic flux plot reconstructed by EFIT on the base of experimental measurements is shown in Fig.6. In the phase of the transition the plasma vertical elongation $\kappa \approx 1.7 \div 1.8$, the plasma volume $V_p \approx 0.42 \div 0.43$ m^3 and the OH power does not exceed P_{OH}=0.3 MW.

The H-mode transition is followed by a formation of a steep gradient in the electron density profile near the separatrix and a flat density distribution inside the plasma volume.

The measured values of the plasma temperature and density together with the plasma voltage

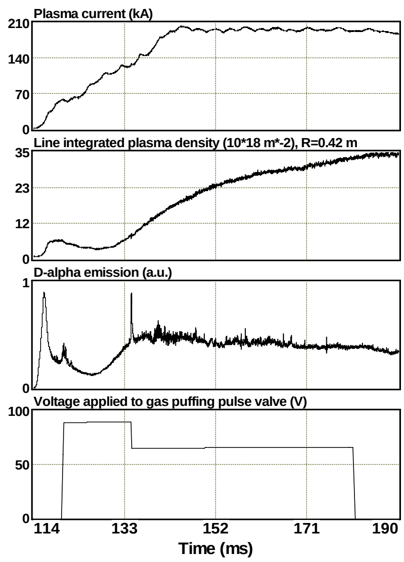


FIG. 5. Time evolution of plasma parameters in OH H-mode. Shot#21987.

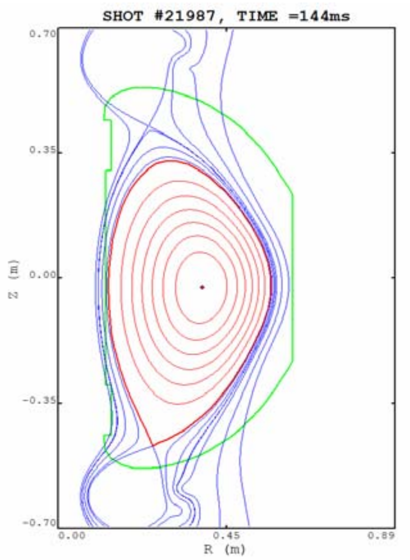
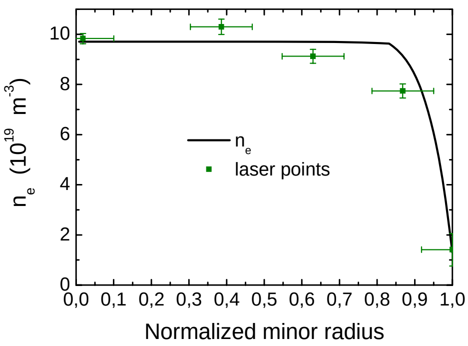


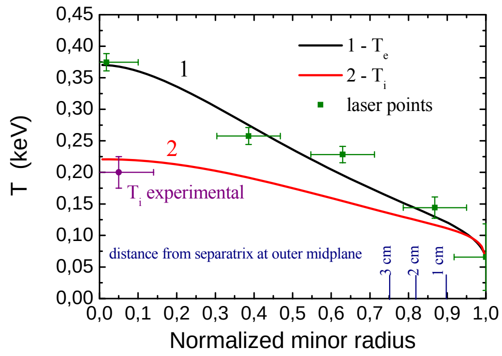
FIG. 6. Magnetic flux plot in the L-H transition phase.

are used as the input parameters for the ASTRA code simulation. The electron temperature and density profiles are measured by the TS diagnostic; the ion temperature is measured by the CX analyzer ACORD-12. The TS diagnostic has 5 spatial channels positioned in the midplane along the major radius within the interval R=0.18÷0.39 m. A special attention was paid to a simulation of transport coefficients.

The transport model consists of the continuity equation for plasma density, the ion and electron energy balance equations and the equation for the poloidal flux solved together with the Grad-Shafranov equation in a real geometry of the spherical tokamak Globus-M. The plasma geometrical parameters are taken from the equilibrium reconstruction performed with the help of EFIT code. The ion energy transport is described by using the neoclassical NCLASS code [10]. The transport coefficients are fitted in such a way that the density and temperature profiles and the plasma voltage corresponded to the experimental data. The measured and simulated density and temperature profiles are shown in Fig.7a and Fig.7b. The respective transport coefficients are shown in Fig.8.



a)



b)

FIG. 7. Measured and fitted electron density and temperature profiles. Shot #21987, time=167 ms.

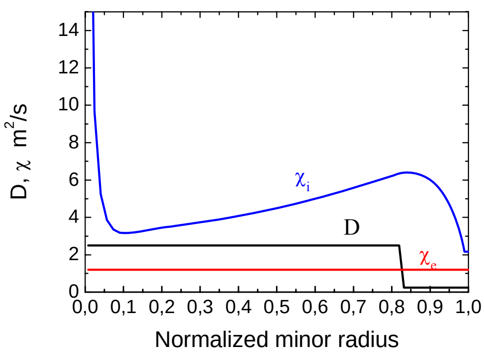


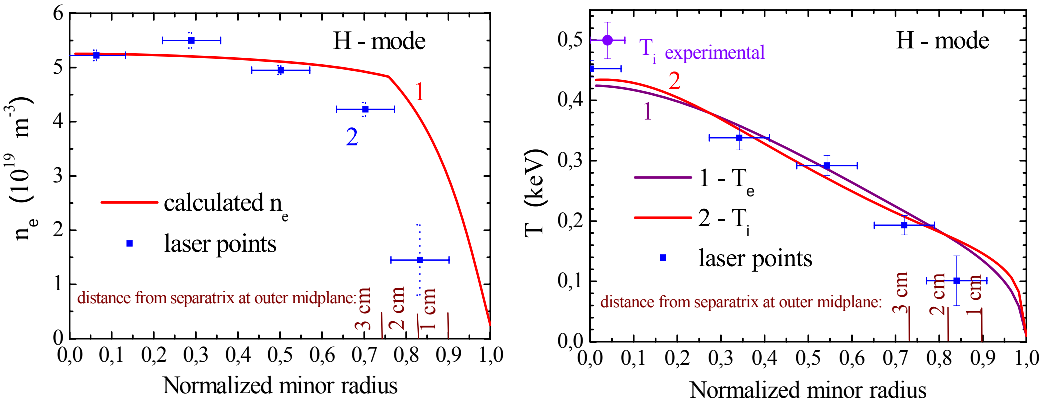
FIG. 8. Simulated transport coefficients. Shot#21987. Time=167 ms.

In the figures the normalized minor radius is defined as a_ψ/a_b, where a_ψ is a half-width of a current flux surface in a midplane and a_b is its boundary value. The calculated particle diffusion coefficient D drops significantly inside the transport barrier near the separatrix compared with its bulk value. The ion heat conductivity χ_i remains neoclassical and no barrier is formed for both ion and electron temperatures. The value of χ_i is approximately by a factor of 4 exceeds the electron heat conductivity χ_e. In the described shot #21987 the plasma thermal energy defined by the ASTRA code on the base of measured and calculated density and temperature profiles is in a good agreement with EFIT energy. Note, that the accuracy of EFIT reconstruction of the plasma energy is sufficiently high [11] due to the large plasma vertical elongation κ≈1.7÷1.8 and the beta factor β_P~0.5 (β_P is the ratio of plasma pressure to the poloidal magnetic field pressure). The plasma thermal energy reaches the value about

2.5 kJ, and the energy confinement time τ_E~8÷10 ms. This value is consistent with the ITER scaling IPB98(y,2) [12].

3.2. H-mode during NBI heating

Stable L-H transition was observed during the injection of deuterium or hydrogen beam into deuterium plasma. Usually the transition was observed in an early injection phase and the H-mode was sustained up to the end of the NBI pulse of 30÷40 ms length. Simulation of a large size beam absorption in the Globus-M plasma is a complicated task and at present the modifications of the beam model for the Globus-M experimental conditions are performing. At the first step of modeling we analyzed the plasma NBI heating with a smaller size beam (characteristic dimensions are 8 cm width and 20 cm height) of a lower power up to 0.5 MW [13]. To some extend it simplifies the simulation and increases the reliability of the results. Typical plasma shot #19518 was selected for the simulation with the NBI power $P_{NBI}\approx0.4$MW, the beam energy $E_{NBI}\approx25$ keV, $I_P\approx0.2$ MA and $B_T\approx0.4$ T. The measured and simulated plasma density and temperature profiles are shown in Fig.9a and Fig.9b.
Similar to OH H-mode the electron density profile has a steep gradient near the separatrix and a flat distribution in the plasma bulk. In the experiment the ion temperature reaches the value of the electron temperature in the plasma center. The simulated T_i profile in Fig. 9b is very close to the measured and simulated T_e profiles. At a lower density the ion temperature



a) b)

FIG. 9. Measured and fitted electron density and temperature profiles. Shot#19518

can exceed the electron temperature. As well as in OH H-mode the transport barrier is absent in the electron and ion temperature profiles.

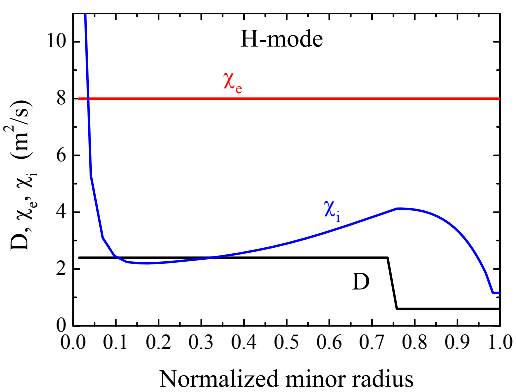


FIG. 10. Simulated transport coefficients.

The transport coefficients are shown in Fig.10. The calculated particle diffusion coefficient D is similar to the ohmic H-mode value. It decreases down to the value of D~0.5÷0.7 m^2/s inside the transport barrier. The ion heat conductivity remains neoclassical. The electron heat conductivity increases almost by an order of magnitude in comparison with the ohmic H-mode. In the modeling the absorbed fraction of the beam power was estimated as 0.2 MW at a comparatively low plasma density. Taking into account the OH power the energy confinement time τ_E~5 ms. As well as in ohmic H-mode this value is consistent with the ITER scaling IPB98(y,2).

4. Discussion and conclusion

The experiments demonstrated an effective NBI and ICRH heating of a small size low aspect ratio plasmas confined in a low toroidal magnetic field. Both methods are based on the plasma heating by means of the energy exchange with supra thermal ions. The neutral beam sizes as well as the characteristic scale of fast ion orbits are comparable with the plasma size. However, the simulations reveal that a noticeable part of fast ions is confined within the plasma volume and a noticeable part of the neutral beam energy can be absorbed at a high density. The CX slowing down energy spectra during NBI are well described by the classic theory due to coulomb collisions. The fast ion first orbit losses together with shine through losses can be large at a low plasma density. In a small size tokamak the experiments aimed at an optimization of the plasma shape and position play an important role. Particularly first simulations of a spatial zone of the confinement of fast ions show that the zone extension depends on the NBI impact radius [7]. Recent experiments [4] demonstrated a tendency to the ion temperature increase when the plasma is slightly (2÷4 cm) shifted along the major radius towards the higher toroidal magnetic field.

The second step also directed to an improvement of the auxiliary heating efficiency is connected with an increase of the toroidal magnetic field which is low in spherical tokamaks due to a restricted room in a torus inner bore. The existed power supply makes possible to increase current in TF coils by a factor of 2÷2.5. This value corresponds to the toroidal magnetic field in the plasma center B_T~0.8÷1 T. But the B_T increase requires the upgrade of magnets including a manufacture of new TF coil inner legs and the central solenoid.

The H-mode transition was achieved at the auxiliary NBI and ICRH heating as well as in the OH regime. In the OH plasma the L-H transition period appeared to be longer in comparison with the L-H transition at the auxiliary heating. A steady state improved confinement phase duration exceeded the characteristic times of plasma transport processes including the time for the diffusion of magnetic field.

The L-H transition is characterized by the steep transport barrier for particle flux in the vicinity of the magnetic separatrix. The ASTRA code simulation reveals that the L-H transition is accompanied by an increase of the electric field shear in the transport barrier region [13], see Fig 11. The neoclassical expression is used for the electric field calculation.

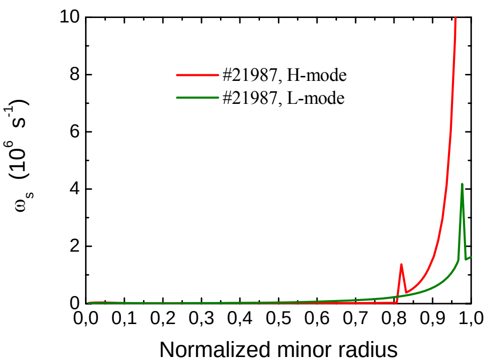


FIG. 11. Poloidal rotation shear.

The critical poloidal rotation shear value is about 10^6 s^{-1}, which is of the order of values typical for ASDEX-Upgrade and MAST [14]. This corresponds to the conception of the turbulent transport suppression.

The L-H transition occurs when the heat power absorbed by ions exceeds the transition power threshold. The threshold power calculated according to the international database [12] P_{LH}, the calculated OH power P_{OH}, the electron-ion power exchange P_{EX}, the power absorbed from NBI by electrons P^e_{NBI} and ions P^i_{NBI} together with average densities and calculated confinement times for both shots for H-mode and L-mode (just before the transition) are summarized in the Table below. Here power is in kW, density is in 10^{19} m^{-3}, time is in ms.

	P_{LH}	P_{ex}	P_{OH}	P^e_{NBI}	P^i_{NBI}	$<n>$	τ_E	$\tau_{IPB98(y,2)}$
#21987 H-mode		~201	~315	0	0	~8.8	~8.7	~8.6
#21987 L-mode	~52	~51	~312	0	0	~3.3	~4.4	
#19518 H-mode		~0	~390	~100	~100	~4.2	~4.6	~5.7
#19518 L-mode	~51	~24	~304	0	0	~1.6	~2	

One can see that in the L-mode the total power exceeds the threshold P_{LH}, given by the data base. The power into the ion channel which plays the key role in the transition is close to P_{LH} for OH plasma. Correct estimates of P^i_{NBI} in the initial NBI phase are complicated presently. The difference in average density and total applied power can qualitatively explain the difference in electron thermal conductivities in H-mode in 2 shots since $\chi_e \sim \frac{1}{\tau_E} \sim \frac{P_{tot}^{0.69}}{n^{0.41}}$ according to ITER scaling.

Acknowledgements

This work is supported by the Russian Ministry of Science and Education, Federal Agency of Russia for Atomic Energy, RFBR grants #07-02-13557, 08-02-00511, 08-02-13537, 05-08-18044.

The authors want to thank Dr. G. Pereversev for ASTRA support and Dr. A. Polevoi for applying NBI solver for small low aspect ratio tokamak parameters.

References

[1] AYUSHIN, B.B., et al., “Study of Plasma Heating in Discharges with Neutral Beam Injection in the Globus-M Spherical Tokamak”, Plasma Phys. Rep. **34** (2008) 81.

[2] SHCHERBININ, O.N., et al., “Numerical Modelling and Experimental Study of ICR Heating in the Spherical Tokamak Globus-M”, Nucl. Fusion **46** (2006) S584.

[3] IZVOZCHIKOV, A.B., et al., Tech. Physics., **37** (1992) 201.

[4] CHERNYSHEV, F.V., et al., “ Fast Particle Confinement Studies in the Globus-M Spherical Tokamak”, Proc. of 35th EPS Plasma Phys. Conf., 9 – 13 June 2008, Hersonissos, Crete, Greece (2008) P-2.097.

[5] PEREVERZEV, G., YUSHMANOV, P.N., Max-Plank IPP report 5/98 (2002).

[6] GUSEV, V.K., et al., Proc. of 34th EPS Conference on Plasma Phys. Warsaw (2007), 2 - 6 July 2007 ECA Vol.**31F**, P-1.078.

[7] CHERNYSHEV, F.V., et al., “Recent Results from CX Diagnostics at Ioffe Institute Tokamaks”, Proc. of 34th EPS Conference on Plasma Phys. Warsaw (2007), 2 - 6 July 2007 ECA Vol.**31F**, P-5.107.

[8] LAO, L.L., et al., Nucl. Fusion **25** (1985) 1611.

[9] SIGMAR, D.J., Varenna Course on Phys. of Plasmas Close to Thermonuclear Conditions, 1, Brussels (1980) 271.

[10] HOULBERG, W.A., et al., Phys. Plasmas **4** (1997)3230.

[11] GUSEV, V.K., et al., “Methods for reconstructing equilibrium plasma configurations in the Globus-M spherical tokamak”, Tech. Phys. **51** (2006) 987.

[12] ITER Physics Basis. Nucl. Fusion **39** (1999) 2175; Nucl. Fusion **47** (2007) S18

[13] SENICHENKOV, I.Yu., et al., “Simulation of L and H regimes for spherical tokamak Globus-M with ASTRA transport code”, Proc. of 35th EPS Plasma Phys. Conf., 9 – 13 June 2008, Hersonissos, Crete, Greece (2008) P-2.097.

[14] ROZHANSKY, V, Modelling of the edge plasma with account of self-consistent electric fields, Conrib. Plasma Phys. **46** (2006) 575