Status of Tokamak T-15MD

P.P. Khvostenko[1], I.O. Anashkin[1], V.A. Belyakov[2], E.N. Bondarchuk[2], O.G. Filatov[2], N.V. Injutin[3], V.A. Krylov[2], I.V. Levin[1], G.E. Notkin[1], A.N. Romannikov[1], M.M. Sokolov[1], A.V. Solopeko[3], A.V. Sushkov[1]

[1]NRC “Kurchatov Institute”, KNTC, Tokamak Department, Moscow, RF

[2] JSC “NIIEFA”, St. Petersburg, RF

[3] GKMP Group, Bryansk, RF

E-mail contact of main author: Khvostenko_PP@nrcki.ru

Abstract. Presently, the Tokamak T-15MD is being built. The magnet system of the Tokamak T-15MD will obtain and confine the hot plasma in the divertor configuration. Plasma parameters are a major radius of 1.48 m, a minor radius of 0.67 m, an elongation of 1.7-1.9 and a triangularity of 0.3-0.4. Tokamak T-15MD will equipped with the auxiliary plasma heating and current drive (Paux =15 - 20 MW) systems. One of the main tasks of experimental study program on T-15MD is the obtaining of physical and technological data needed both for ITER project support and fusion neutron source creation. The manufacturing of the Tokamak T-15MD magnet system and of vacuum chamber shell was completed. At present time, the preliminary assembly of the magnet system is carried out at the plant in Bryansk. The disassembly of the superconducting tokamak T-15 have been finished in august, 2016 and T-15MD assembling must begin in NRC “Kurchatov Institute” in 2017. Status of T-15MD engineering systems modernization is presented too.

1. Introduction

The main trend in the development of society in the 21 century is the continuous growth of energy consumption associated with both population growth and the increasing energy needs of third world countries, which is essential for their development. The possibilities of the growing demand for energy by means of fuel expense of organic origin, and of renewable sources are limited due to their natural limits. The basic strategic directions of the efforts in controlled nuclear fusion (CNF) research in the Russian Federation are the support and participation in the ITER project and the participation in the development of the demonstration fusion reactor DEMO as the pure thermonuclear energy sources. The development and creation of stationary fusion neutrons source based on tokamak is the real way for the solution of these problems. The ideology of neutron sources for burning actinides and developments of fuel for nuclear power in Russia is considered one of the promising to achieve faster return on CNF research and tangible demonstration of the energetic capabilities of fusion. Requirements for plasma parts and neutron fluxes for hybrid reactors less than a purely thermonuclear reactor, which can significantly reduce the time needed for their implementation.

At present time, in NRC “Kurchatov Institute” within the Federal Target Program “Nuclear energy-technologies of new generation for period 2010 - 2015 and to the prospect until 2020” the tokamak T-15MD [1] with divertor configuration of plasma column is built and the modernization of technological systems is accomplished.

2. Goals of T-15MD

Experimental study program on tokamak T-15MD is goaled on the obtaining of physical and technological data for both ITER support and creation of the fusion neutron source. The program includes decision of the following tasks: possibility of achieving the high values of beta as way to the reduction of fusion reactor cost; control by the plasma current and pressure profiles; study of the possibility of obtaining of both high beta and plasma density in steady-state discharge with the completely non- inductive current; on-line control of stability, equilibrium, heating and confinement of high-temperature plasma; study of plasma interaction with different materials, including graphite, tungsten and lithium. Tokamak T -15MD will be used as the test bed, on which will be mastered such systems, as the stationary injectors of neutral particles, ICR, ECR and LH plasma heating systems, will test materials and the technologies for the first wall and divertor.

Installation will equipped with the auxiliary plasma heating and current drive (P aux = 15 - 20 MW) system, which allow also the simultaneous achievement of high plasma temperature (Ti ~ Te ~ of 5-9 keV) and plasma density (ne ~ 10^{20} m^{-3}) with pulse duration up to 30 s. This system includes a neutral beam injectors (3 co-injectors of 2 MW/ 75 keV H^0 each), electron cyclotron resonance heating (7 gyrotrons of 1.0 -1.5 MW each, f =110-120 GHz with a possibility both of second harmonic ECR and electron Bernstein wave heating), ion cyclotron resonance heating (3 antennas of 2 MW each including the possibility of the helicon wave generation), low hybrid heating and current drive (grill of 4 MW power, f=2.45 GHz).

3. Magnet System

In 2015, at plants in St.-Petersburg and Bryansk the manufacturing of the all elements of electromagnetic system (EMS) of the tokamak T-15MD was completed. EMS should obtain and confine of the hot plasma in the different divertor configurations. The EMS consists of a toroidal winding, a poloidal magnet system and supporting structures [2, 3]. The toroidal winding consists of 16 toroidal field (TF) coils and of inter-coil structure. The poloidal magnet system includes a central solenoid (CS), poloidal field (PF1-6) coils and horizontal field (HF) coils.

3.1. Toroidal Winding

The toroidal winding consists of 16 D-shaped coils (Fig.1) forming the arched structure. The toroidal magnetic field at the plasma axis is 2 T. The level of ripples at the outboard plasma boundary is approximately 0.8 %.

3.2. Poloidal Magnet System

The poloidal magnet system generates the divertor with single null and double null magnetic configurations and the plasma shape with elongation k_{95} = 1.7-1.9 and triangularity δ_{95} = 0.3-0.4. The poloidal magnet system includes the central solenoid, consisting of three coils, the six poloidal coils and the four horizontal coils.

3.2.1. Central Solenoid

The central solenoid consists of three separated coils (of 151, 449 and 151 turns). Each coil is wound with two parallel hollow conductors of trapezoidal form (dimensions are 14.8 x15.5 × 20 mm^2 with ∅8 mm ID), made of silver-copper alloy. Coils are charged by independent power supply systems. The magnetic flux swing in central solenoid Ψcs is approximately 6.5 Wb ($I_{cs} = \pm 40$ kA). The central solenoid, manufactured at a plant in Bryansk is shown in Fig. 2.

FIG.1. Coil of toroidal field winding at plant in Bryansk.

FIG.2. Central solenoid.

3.2.2. Poloidal Field Coils and Horizontal Field Coils

PF (1-6) coils must compensate for stray magnetic fields on the breakdown stage, maintain the equilibrium of the plasma column at the stages of current ramp-up, maintain the ramp-down of the plasma current and create the limiter and the different divertor configurations. PF coils are placed outside the toroidal winding and are fastened to the TF coil cases. For fast plasma position control the four active control coils (HF) are placed around the torus in the space between the vacuum vessel shell and the toroidal winding. All coils are fed by independent current sources and are cooled by distilled water. PF (1-6) coils and HF coils are shown in Fig.3.

FIG.3. Poloidal field (PF1-6) coils and HF coils.

4. Preassembly of Electromagnetic System

At present time, a preassembly of the electromagnetic system is carried out at plant in Bryansk (GKMP Group). The purpose of the preassembly is check the conjunction of all elements of EMS among themselves, the installation of intercoil mechanical structures. The toroidal field coils after its establishment should not create fields, across the main toroidal field 2 T higher than 5 G. This requirement means that the maximum deviation of the top of the coil on its supports must not exceed one millimetre.

The inter-coil mechanical structure is attached to coil cases by welding. Initially, all four series of inter-coil mechanical structure (lower, lower intermediate, upper intermediate, upper) have been welded with point contact. Final welding has been performed after verification the installation of inter-coil structures. Assembling the toroidal magnet system, consisting of 16 TF coils, with inter-coil mechanical structure is shown in Fig. 4.

To install a four framed form active control coils placed around the torus in the space between the vacuum chamber shell and the toroidal winding, one toroidal field coil has been removed. The position of the horizontal field coils relative to the TF coils controlled by laser level. Mount of HFC was carried out with the help of brackets that are welded to the to TF coil cases. HF coils were clamped between them. Placement of HF coils inside toroidal winding is shown in Fig. 5.

FIG. 4. TF coil assembly with inter-coil mechanical structure.

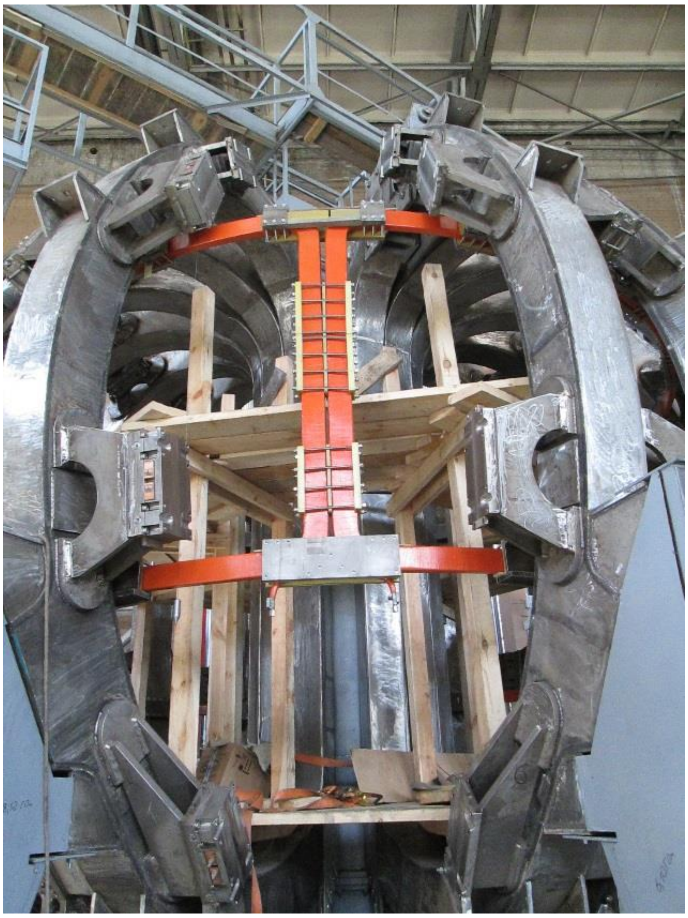

FIG. 5. Placement of HF coils inside toroidal winding.

5. Vacuum Vessel

Vacuum vessel has sixteen equatorial rectangular ports and 136 ports of various diameter intended for diagnostic and technological purposes. All flange connections of ports, except equatorial ones, performed on metal seal of conflat type. The inner surface of the vessel is equipped with ribs, which sets out the graphite tiles. On external and internal contour on the inner surfaces of conical shells and cylinder, the passive stabilization copper turns protected by graphite tiles will be installed. Upper and lower divertors must accept a heat fluxes up to 10 Mw/m^2 without loss of efficiency. The baking system is performed by ohmic heaters of cable type and provides the vessel shell heating up to 220°C and divertor tiles up to 400°C. The outer surface of the vessel is closed by thermal insulation. The temperature at the outer insulation surface will no more than 40°C. A large number of sensors of electromagnetic diagnostics placed inside vessel. Shell of vacuum vessel is shown in Fig. 6. Vacuum tests will be carried out in October 2016.

FIG.6. Vacuum vessel shell.

6. Engineering systems

After introduction into operation in 1988 of the T-15 superconducting tokamak the technological equipment has developed its own resource and it took its replacement by the contemporary one. Modernization of engineering and technological systems has improved the technical characteristics of the systems and their reliability and automation of technological processes will increase the trouble-free operation and ensure significant reduction of staff.

6.1. Vacuum pumping system

Four turbo molecular pumps provide high vacuum pumping of vessel with total productivity ~ 10 m^3/s (H_2) and two cryogenic pumps have the productivity 4 m^3/s (H_2) each. The pumps removed from the installation up to the distance of 6 m to avoid their damage because of the appearance of the stray magnetic fields at operation with plasma. All vacuum pumping equipment was tested with using of a control system.

6.2. Water cooling system

The magnet system, in-vessel components, the turbo molecular pumps, equipment of auxiliary plasma heating systems are cooled by distilled water with pressure 0.5 MPa and 1.0 MPa. The heated distilled water is cooled in heat exchangers by river water. The total mass flow rate of distilled water is 2000 m^3/h and river water is 1500 m^3/h. The equipment for water cooling system was supplied and mounted.

6.3. Power supply system

Total power consumption during the pulse with plasma current 2 MA and additional plasma heating of 20 MW will be 300 MVA. Reconstruction of substations No. 745 (110/10 kV/1 kV) and No. 110 (110/10 kV/0.4 kV) will be completed in the 2017-2018. Eighteen thyristor converters and transformers for power supply system of toroidal winding and the central solenoid will be purchased in 2017-2019.

6.4. Information and control system

Information and control system for tokamak T-15MD [4] is modular, scalable system that is intended to use on big physical research device like Tokamaks. Core of this system is Central Control System that coordinates and controls the main subsystems groups:

- Technological subsystems, including special systems that provide the operation of tokamak itself (power supply system, vacuum system, water cooling system, cryogenic system etc.). These subsystems are based on Schneider Electric controllers and Wonder Ware software;
- Fast problem-oriented subsystems, including interlocks, plasma control, synchronization and coordination systems, work in a real-time scale during experiments with plasma and which ensure the desired parameters and characteristics. These subsystems are based on NI platform and partially on VME;
- Informational and measurement complex for data acquisition, data storage and processing of experimental data. Complex based on NI platform, MS Windows and MS SQL.

At the present, the Data Centre (Fig. 7) created, equipment installation of main systems, testing and verifying some subsystems are held to implement physical commissioning installation in 2018-2019.

7. Time schedule

The schedule for the commissioning of tokamak T-15MD is the following: 2016-vacuum test of vessel shell at the plant in St. Petersburg, delivery the vacuum vessel shell to the plant in Bryansk for joint preassembly with the electromagnetic system, delivery of separated elements of electromagnetic system and vacuum vessel to NRC "Kurchatov Institute"; 2017-assembling the tokamak T-15MD, reconstruction of substations No. 110 and no. 745; 2018-adjustment of individual technological systems in conjunction with the tokamak T-15MD, a physical start-up.

FIG. 7. Data Centre.

8. References

[1] Azizov, E.A., Belyakov V.A., Filatov O.G., Velikhov E.P. and T-15MD Team. Status of Project of Engineering-Physical Tokamak // 23rd IAEA Fusion Energy Conference, Daejeon, Korea Rep. of 11-16 October 2010, FTP/P6-01.

[2] Bondarchuk, E.N., Azizov E.A., Alekseev A.B.,...Khvostenko P.P. et al. Engineering Problems of Tokamak T-15 Electromagnet System Reconstruction // IEEE Transactions on Applied Superconductivity, 2012, Volume: 22 , Issue: 3, Page(s): 4201604.

[3] Khvostenko P.P., Azizov E.A., Alfimov D.E. et al. The magnet system of the Tokamak T-15 upgrade // Fusion Engineering and Design, 2015, vol. 98-99, 1090-1093.

[4] Motckin Y.Ya., Sokolov M.M., Khvostenko P.P. Information and control system for experimental fusion device “Tokamak T-15” // MKA:BKC, 2015, №3, 29-37.