Status of Project of Engineering-Physical Tokamak

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Abstract. Status of project of Engineering-Physical Tokamak, named T-15MD, is outlined. The main goals of tokamak T-15MD are: long pulse, non-inductive current drive regime realization in plasma configuration with divertor at aspect ratio $A \sim 2.2$ and plasma elongation $k \sim 1.7$-1.9. By most important component of project is the accumulation of the database, necessary for the transition to the following step – thermonuclear neutron source on the basis of tokamak.

1. Introduction

Engineering-Physical Tokamak, named T-15MD is intended for solution of broad spectrum of physical problems and technology development, necessary for substantiation of fusion power plant parameters. Tokamak T-15 MD must become the physical prototype of neutron source for transmutation and fuel production.

Tokamak T-15 MD is planned as contemporary compact tokamak with the flexible ITER-like configuration of magnetic field, which makes it possible to obtain the form of plasma column with the aspect ratio in the range 2.2-3, with the non-inductive current drive and the long pulse, and also with the control capability of form and of the parameters of plasma column in the real time.

The base parameters of tokamak T-15MD are given in Table 1.

TABLE 1: BASE PARAMETERS OF TOKAMAK T-15MD

Plasma current I_P, MA	2
Toroidal magnetic field at plasma axis B_t, T	2
Major radius of torus R, m	1.48
Aspect ratio A	2.2
Elongation, k_{95}	1.7 – 1.9
Triangularity, δ_{95}	0.3 – 0.4
Pulse duration Δt, s	10
Plasma density, 10^{20} m^{-3}	0.5
Plasma temperature Ti(0), Te(0), keV	4-5
Normalized beta β_N	1.5
Power of auxiliary plasma heating P_{AUX}, MW	13

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The start of discharge and the plasma current ramp-up is proposed to perform by an induction method in combination with ECR heating at the level 200 kW. After reaching of the base value of plasma current transition to non-inductive current drive must be provided.

Physical tasks of T-15MD include investigation of: electron and ion transport; plasma disruptions; plasma turbulence, ITER-like (in non-dimensional parameters) discharges for specification of non-dimensional dependences; reaching the regime of the prolonged non-inductive plasma burning with the high parameters (β_N, $H_{y,2}$, q_{95}, k, δ, n_e/n_G); technological problems.

For the solution of these problems in T-15MD it is necessary:
- to use auxiliary heating systems, elaborated for T-10 and T-15 tokamaks with total power Paux ≥ 15 MW both for heating of ions (NBI, P_{NB} = 8-10 MW) and of electrons (ECRH , P_{ECRH} = 8 MW, f = 110-120 GHz);
- to create ICR and LH plasma heating systems;
- to provide current drive generation by NBI, ECCD, LHCD and control of plasma profiles;
- to develop modern complex of diagnostics.

T-15MD will be built in Kurchatov Institute with use of infrastructure of T-10 and T-15 tokamaks. This fact leads to limitation of possibility of power and water consumption. Moreover it is important to join vacuum chamber and electromagnetic system of T-15MD with existing systems of plasma heating (ECRH and NBI systems for T-10 and T-15).

2. Basic scenarios of the tokamak T-15MD

Scenario of plasma discharge on this stage of work is traditional: plasma breakdown is organized at toroidal field plateau, position of the breakdown region is near inboard wall, plasma current ramp-up at 1.5 – 2 MA/s, auxiliary plasma heating is switched on at plasma current plateau.

Plasma initiation analysis is done by TRANSMAK code and gives following preliminary results:
- plasma start with small breakdown region is preferable (a = 0.25 m, R = 1.06 m);
- maximum initial poloidal flux capacity in the central solenoid (if stray field at the breakdown region is ≤ 1 mT) is Ψ_0 = 3.45 Wb for given limits for power supply;
- flux capacity at the end of plasma initiation stage (when plasma current reaches 40 – 50 kA) is ≈ 3.25 Wb.

Parameters both of Ohmic scenario and of scenario with auxiliary heating with pulse duration Δt = 4 - 5 s (ASTRA code simulations) are given in *Figs.1,2.* As follows from data on *Fig.1*, for Ohmic scenario resistive voltage is ~ 0.7 – 0.8 s, energy confinement time ~ 200 ms and averaged temperature ~ 1.5 keV. L-H transition power is $P_{L\text{-}H}$ ≈ 1 – 1.5 MW.

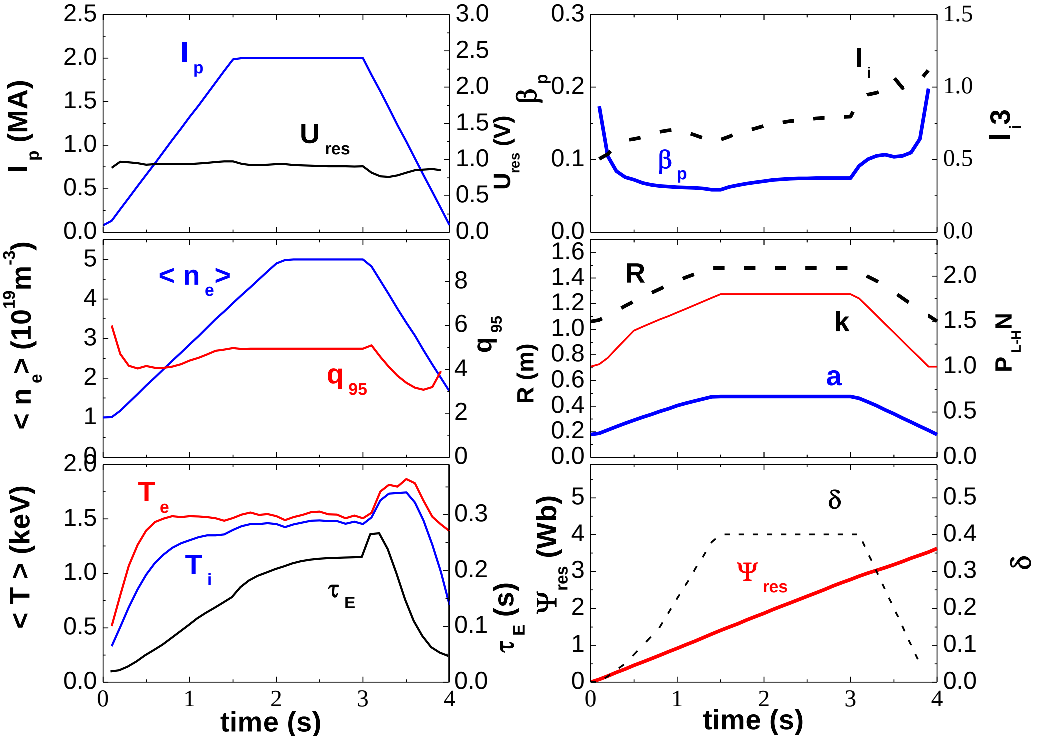

Fig.1. T-15MD: evolution of plasma parameters for Ohmic scenario: plasma current I_P and resistive plasma voltage U_{res}; poloidal beta β_P and plasma internal inductance l_i; averaged plasma density $<n_e>$ and safety factor q_{95}; major and minor radiuses R, a and plasma elongation k; electron and ion average temperatures T_e, T_i and energy confinement time τ_E; plasma triangularity δ and poloidal resistive flux Ψ_{res}.

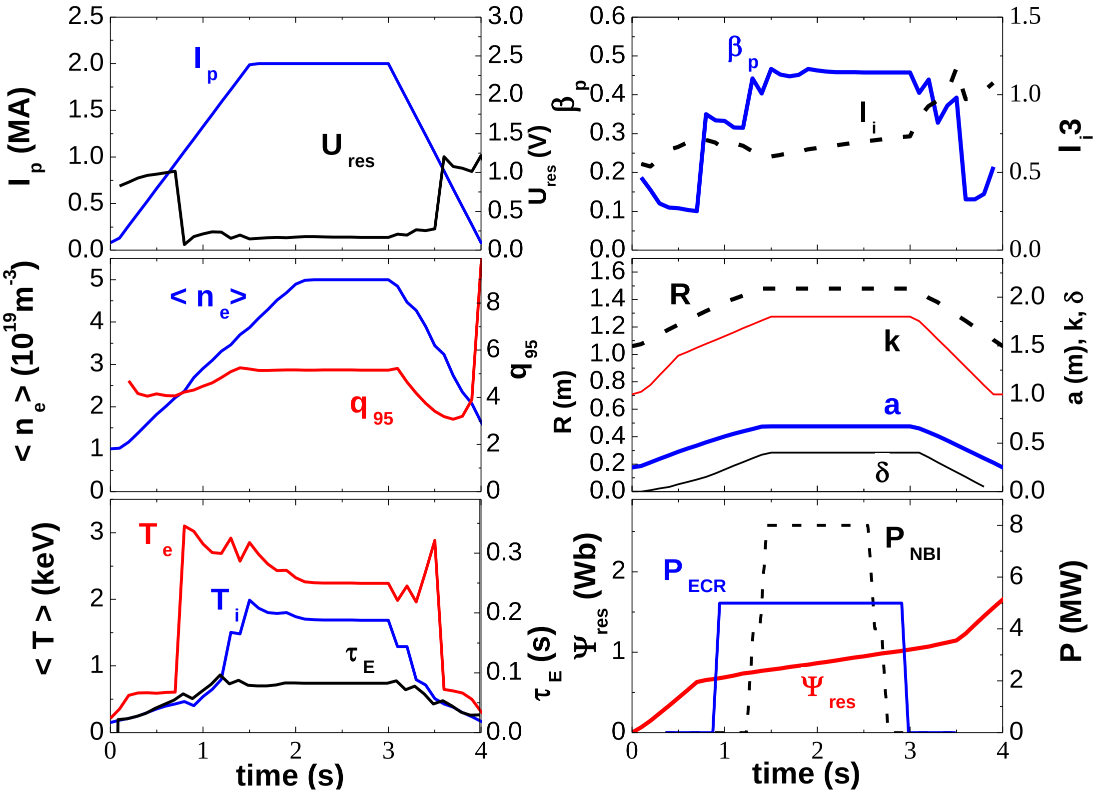

Fig.2. T-15MD: evolution of plasma parameters for scenario with additional heating: Level of heating power by ECRH and NBI are shown in the last subfigure

Scenario with additional heating (total power 13 MW: 5 MW ECRH + 8 MW NBI, *see Fig.2*) at $n_e = 5 \cdot 10^{19}$ m^{-3} allow to increase average plasma temperature up to 3 keV (5 keV at the centre) and decrease loop voltage to 0.2 V.

3. T-15MD Design

General view of T-15MD tokamak and position of main subsystems are shown in *Fig.3*.

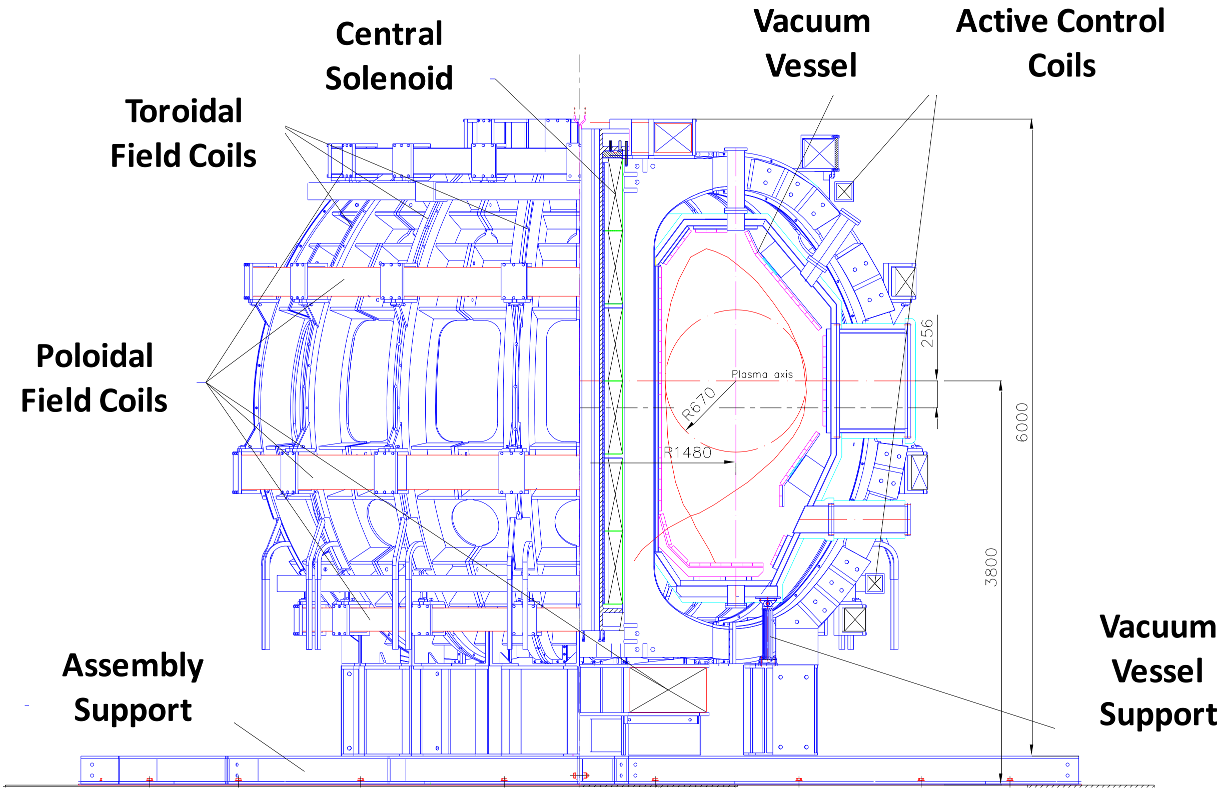

Fig.3. General view of T-15MD tokamak

3.1 Electromagnetic system

T-15MD electromagnetic system is “warm”.

3.1.1. Toroidal field coils

Toroidal magnetic field is created by 16 D-form coils (30 turns in each coil). The value of toroidal magnetic field at major radius 1.48 m is 2 T. Level of ripples at outboard plasma boundary is 0.6 – 0.8 %. Toroidal field coils are multi-turn, non-sectional. Assembly view of toroidal field systems with inter-coil structure is shown in *Fig.4*.

3.1.2.Poloidal field coils

Poloidal field systems consists of central solenoid and six PF coils (PF1 – PF6). Central solenoid consists of three separate coils (it is possible to increase number of CS independent coils up to six, as shown in Fig.3). Positions of poloidal field coils are given in Table 3 (*see also Fig.3*).

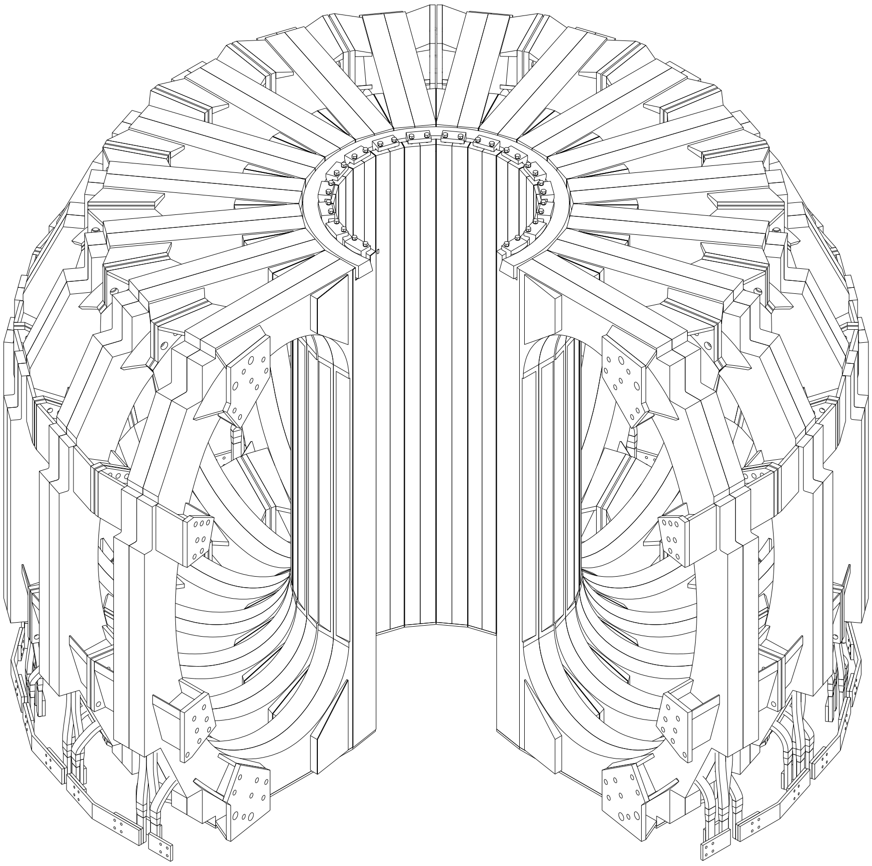

Fig.4 Toroidal assembly with inter-coil structure

TABLE 3: POSITION OF POLOIDAL FIELD COILS

	Rs, m	Zs, m	ΔR, m	ΔZ, m
CSU	0.335	1.335	0.130	1.262
CSC	0.335	0.000	0.130	1.262
CSD	0.335	-1.335	0.130	1.262
PF1	0.880	2.218	0.222	0.192
PF2	2.328	2.044	0.210	0.188
PF3	2.911	0.934	0.124	0.264
PF4	3.210	-0.866	0.352	0.380
PF5	2.541	-2.230	0.156	0.198
PF6	0.887	-2.756	0.494	0.188

View of equilibrium plasma configuration (t = 1.6 s) for plasma current I_P = 2 MA is shown in *Fig.5*.

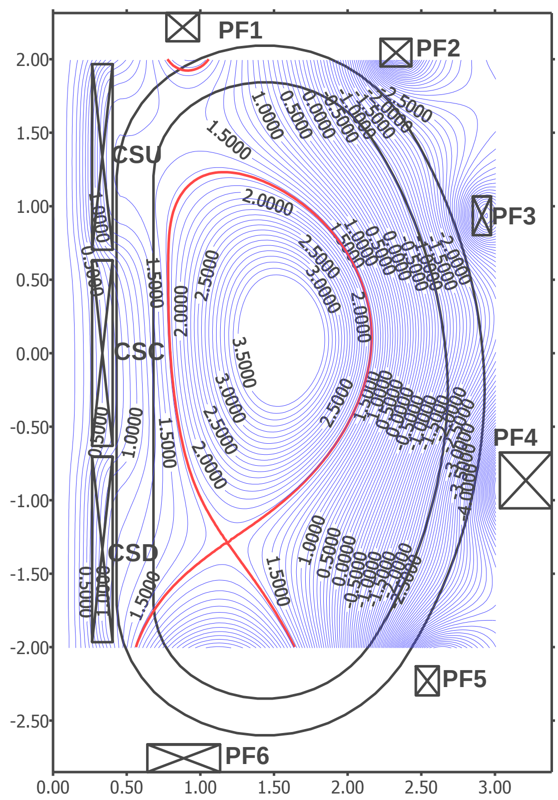

Fig.5. Plasma equilibrium at t = 1.6 s (I_P = 2 MA).

Evolution of PF&CS coil currents for Ohmical plasma scenario (scenario of physical parameters see in *Fig.1*) is given in *Fig.6*.

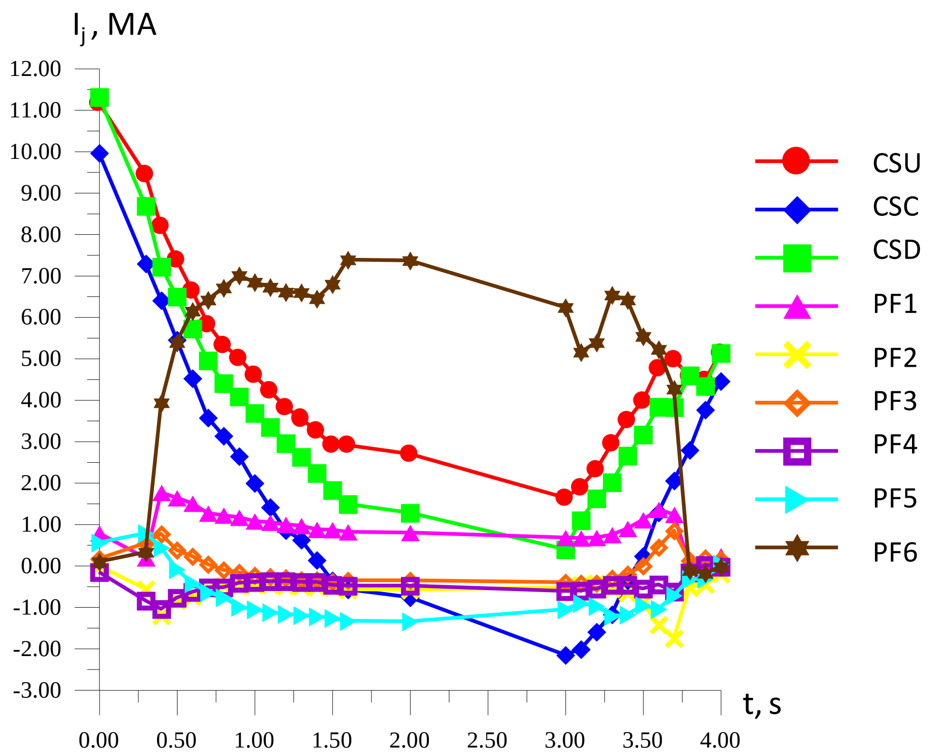

Fig. 6. Evolution of PF&CS coil currents for Ohmic scenario

3.2 Vacuum chamber

Vacuum chamber includes:

- shell with ports, first wall, passive stabilization coils and supports;
- systems of electromagnetic and technological diagnostics;
- baking and cooling systems;
- divertor;
- vacuum pumping systems.

General view of vacuum vessel with ports, passive stabilization coils and divertor is shown in *Figs.7,8.*

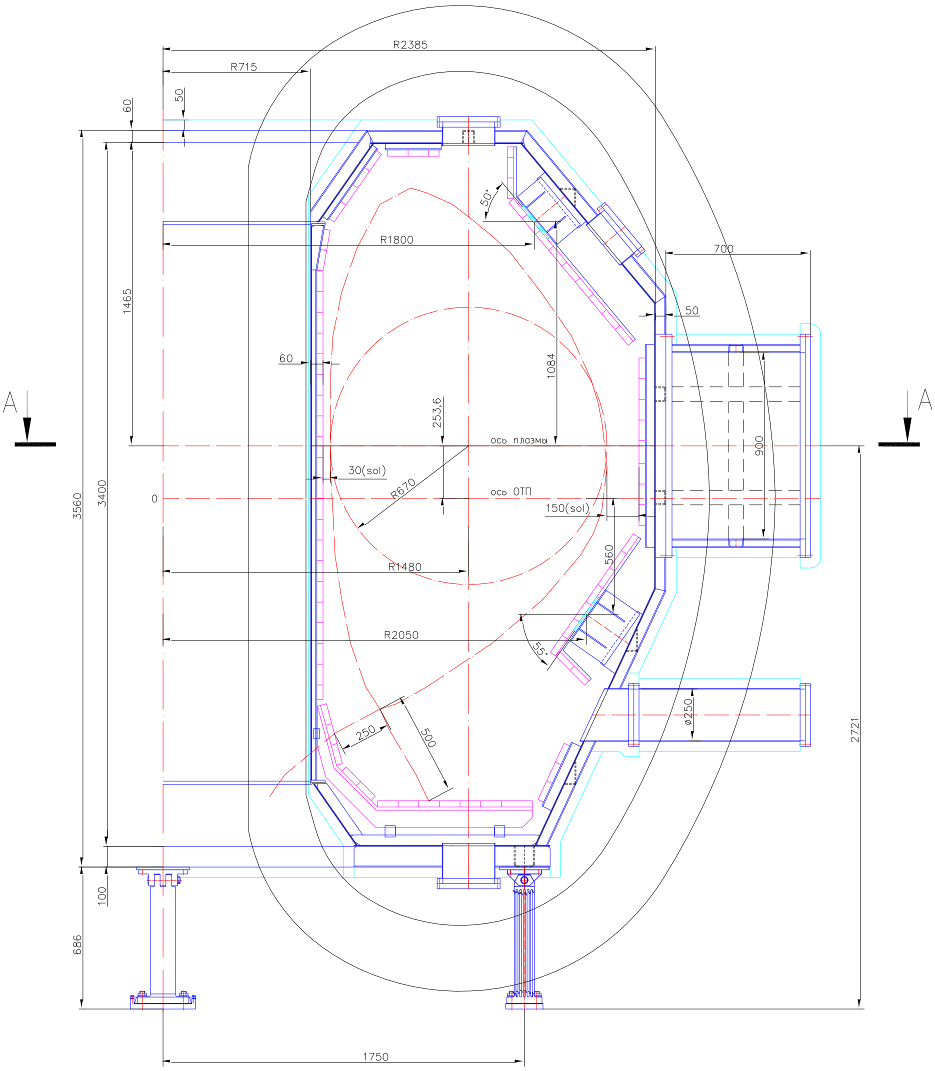

Fig.7. Cross section of vacuum chamber

Dimensions of equatorial ports allow realize the tangential input of NB injection (*see Fig.7*). Ports in bottom part of vacuum chamber are intended for divertor diagnostics, for divertor plates maintenance and for vacuum pumping.

Ports in bottom part of vacuum chamber are intended for diagnostic purposes.
Vacuum vessel baking is proposed at temperature 220 ± 20^0 C.

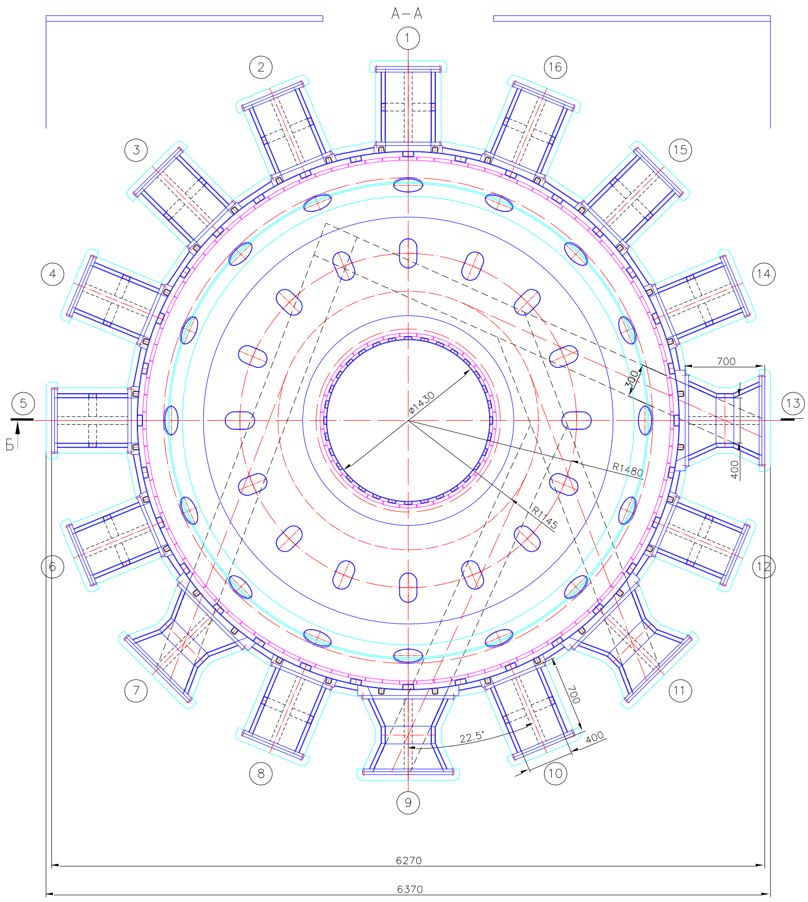

Fig.8.Top view of vacuum chamber

Divertor design is now at initial stage. Divertor construction consists of plates located near inner and outer separatrix. Distance from X-point is (in cross section) 0.5 m to outer separatrix and 0.25 m – to inner one.

Divertor plates are cooled by water. Construction of divertor plates allows replacing theirs under exploitation.

4. Conclusions

Status of project of Engineering-Physical Tokamak, named T-15MD, is outlined. The main goals of tokamak T-15MD are: long pulse, non-inductive current drive regime realization in plasma configuration with divertor.

Tokamak T-15MD is planned to have following main parameters: major plasma radius 1.48 m, minor radius 0.67 m, aspect ratio 2.2, elongation 1.7 – 1.9, triangularity 0.3 – 0.4, toroidal magnetic field at plasma axis 2 T, plasma current 2 MA, magnetic flux swing in central solenoid ~ 6 Wb, total power of auxiliary plasma heating 10-15 MW, pulse duration 5 – 10 s.

Tokamak T-15 MD must become the physical prototype of neutron source for transmutation and fuel production. Tokamak T-15MD is proposed as center of fusion investigations in Russian Federation, which will unite scientific and technical potential of different research institutes.