UDC 533.9

Discharge Scenario for T-15M tokamak design project.

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1. Introduction. T-15M tokamak design project.

The work for the design project new tokamak T-15M creation is carried out during 1999-2000 years by NFI, RRC "Kurchatov Institute", Efremov Institute, TRINITI and Lomonosov MSU. The experimental device T-15M is intended for solution of the broad spectrum of the physical problems and evolution of the technologies which are necessary for reliable design of the power nuclear fusion energy reactor. T-15M is necessary for further development of the experimental investigations which are carried out on tokamak T-10, T-15M, T-11M, TUMAN-3 and GLOBUS-M. This device should provide base center for the Nuclear Fusion investigations in Russia which will consolidate scientific and technical potential of the Russia researcher laboratories. The new tokamak will allow to keep and develop technical and engineering, experimental and theoretical plasma physic and nuclear fusion devices.

The main purposes of T-15M are defined by the necessity to provide maximal integration into international research program for creation thermonuclear reactor (ITER, DEMO). Solution of the problems to support international project ITER and allowing wide research program by controlled nuclear fusion in Russia Federation.

The T-15M experimental program cover a wide range of researches by controlled nuclear fusion and include the following problems:

- The possibility of operation by high β_N, high density and temperature simultaneously in order to decrease cost of the tokamak-reactor.
- The plasma current and pressure control in order to increase of the β_N and energy retention time τ_E.
- Operation in H-mode with internal and external transport barrier.
- Improvement of the neoclassical and ideal limits.
- Realization of the regimes with high β_N and n_e in conditions of the stationary discharge with totally non-inductive current.
- The optimization of the divertor and analysis of the effects of the peripheral plasma on the global characteristics of the plasma.
- Analysis of the vertical plasma stability.
- Investigation and algorithmization of feedback system for stability of the H-mode.

Solution of this problems which can not be solved completely on other devices during construction T-15M (near 5-6 years) is necessary for elaboration of the advices for building ITER and tokamak-reactor.

It is necessary for solution of the referred above problems:

- The auxiliary heating systems developed for T-10 and T-15 with power not less the P_{aux} = 15 MW, which provide heating of both ions (neutral beams injection system, P_{NBI} = 8-10 MW), and electron (gyrotron complex, P_{ECRH}=8 MW, f = 110-120 GG) components should be used. The auxiliary plasma heating system with existing ion-cyclotron and low-hybrid microwave should be created.
- The possibility of the generation of the non-inductive current equal to plasma current (NBI, ECCD, LHCD) and control by plasma current profile should be provided.
- The equipment of the device by the modern diagnostic complex must be envisaged.

Tokamak T-15M will be constructed into NFI RRC "Kurchatov Institute". T-15M should be fitted into NFI infrastructure for maximal use of the manufactured equipment from T-10 and T-15 devices. Electromagnetic system configuration should be similar to ITER configuration. The T-15M poloidal field system should provide the operation with poloidal divertor plasma configuration. T-15M should be constructed during 5 years from beginning of the designing. The cost of the T-15M basic systems is estimated at 6 millions dollars.

The further T-15M modernization foresee an increase of the auxiliary heating systems. It allow to realize the regimes with limit β_N values.

The T-15M should provide investigation of the plasma with parameters are shown in the Table.

Problems of Atomic Science and Technology. 2000. N 3. Series: Plasma Physics (5). p. 25-27

Table. The main T-15M parameters

Plasma current I_{pl}, MA	1.7
Aspect ratio, À	3.1
Large radius $R_{î}$, ì	1.55
Minor radius a, ì	0.5
Elongation k_{95}	1-2
Triangularity δ_{95}	0-0.5
Toroidal magnetic field at R_o -B_T, T	$2 \le B_T \le 2.5$
Duration of the plasma current plateau $\Delta t_{plateau}$, s	≥ 5
Auxiliary heating power P_{aux}, MW	≥ 15
Duration of the auxiliary heating, s	4
Plasma density n_e, 10^{20} ì $^{-3}$	1
Plasma centre temperature, keV	5
Total poloidal flux swing $\Delta\Psi_{cs}$, V.s.	≈ 6
Plasma shape	SN

The several configurations of the poloidal field systems was considered during design of the T-15M project. The algorithm and results of the calculations one of discharge scenario of the T-15M are presented in this paper.

2. Modelling of the discharge scenario.

Under discharge scenario for tokamak plasma we will understand the following definition. Spatial position of the poloidal field coils (PFC) are prescribed. The parameters: *plasma loop voltage (U_{pl}), plasma current (I_{pl}), poloidal beta (b_{pl}), major plasma radius (R_{pl}), minor plasma radius (a_{pl}), plasma ellipticity (l_{pl}) and plasma triangularity (d_{pl})* (*) are preassigned as the functions of time. It is required to define the magnitudes of the currents in the PFC, so that the plasma has parameters (*) in the every time point.

We examine the ohmic heating regime for T-15M.

In the initial time point we must compensate the scattered poloidal magnetic field from central solenoid (CS) with help currents in the other PFC (PF) Fig.1. In order to find initial values of PF currents and plasma boundary poloidal magnetic flux (ψ_{pl}) let assign maximal CS current and small plasma current I_{pl} = 10 kA. We examine that break down takes place on the external vessel side at R_{pl} *=1.9 m*, and radius of the break down region a_{pl} *= 0.2 m*. At the initial time point the plasma equilibrium is provided with help limiter. The PF current and ψ_{pl} are found with the help of solution equilibrium problem. It is the initial time point of the discharge.

As we assigned the plasma loop voltage (U_{pl}), and found the initial values of the poloidal magnetic flux on the plasma boundary ψ_{pl}^0 we can find the time dependence of $\psi_{pl}(t)$ by solving the differential equation with initial conditions.

$$\frac{\partial \psi_{pl}}{\partial t} = -U_{pl}, \quad \psi_{pl}(0) = \psi_{pl}^0$$

The numerical code is used for the plasma equilibrium calculations. The following 2D Shafranov equilibrium problem for a tokamak is solved:

$$\Delta^*\psi = -0.8\pi^2 \cdot r \cdot \begin{cases} j_\varphi(r,\psi), & \psi \ge \psi_{pl} \ (S_p) \\ \sum_{j=1}^{N} I_j \cdot \delta(r-r_j)\cdot\delta(z-z_j), & \psi < \psi_{pl} \ (S_\upsilon) \end{cases},$$

$$y\big|_\infty = 0,$$

where S_p, S_V are the plasma and vacuum regions,

y is the poloidal magneticflux: $B_r = -\frac{1}{r}\frac{\partial\psi}{\partial z}$, $B_z = \frac{1}{r}\frac{\partial\psi}{\partial r}$, B - is the poloidal magnetic field,

r, j, z are the cylindrical coordinates with z-axis directed along the main torus axis,

$j_j(r,y)$ is the toroidal plasma current density:

$$j_\varphi(r,\psi) = I_{pl}\left[r\frac{\beta_{pl}}{N_P} + \frac{(1-\beta_{pl})}{rN_I}\right]\cdot(\psi-\psi_{pl})^\alpha,$$

$$N_p = \int_{S_p} r\cdot(\psi-\psi_{pl})^\alpha ds, \quad N_I = \int_{S_p} \frac{1}{r}\cdot(\psi-\psi_{pl})^\alpha ds,$$

$$\beta_{pl} = \frac{8\pi\langle nT\rangle}{B_p^2},$$

N_p, N_I are the normalizing constants, I_{pl} is the total plasma current,

the parameter α determines the plasma current density profile (internal inductivity l_i),

I_j, r_j, z_j are the currents in the poloidal field coils and their centers coordinates, $\delta(r)$ is the Dirac delta-function, $\Delta^* y \equiv r\cdot\frac{¶}{¶r}\left[\frac{1}{r}\frac{¶y}{¶r}\right] + \frac{¶^2 y}{¶z^2}$.

At any time point t we solve the 2D Equilibrium Problem and find the currents in the PFC thus that the plasma has the parameters (*) and $\psi_{pl}(t)$. We assumed that the plasma current has linear profile on magnetic surfaces (α=1), β_{pl} = 0.1 is constant.

The space distribution of the PFC, passive coils (PC) and the plasma boundary at different time point are shown in Fig.1.

The time dependencies the assigning plasma current (a), calculated central solenoid current (b) and poloidal field current (c) are shown in Fig.2.

The time dependencies of the assigning: U_{pl} (a), R_{pl} (b), a_{pl} (c), λ_{pl} (b), δ_{pl} (c) and calculated: ψ_{pl} (c), l_i (b) parameters are shown.

In this scenario on the initial discharge stage the plasma boundary is formed by the limiter. The needed separatrix plasma configuration is created at the beginning plasma current plateau and kept till the end of the stationary state. At initial stage the plasma is kept near circular ($\lambda_{pl} \sim 1$) so that decay index was positive during maximal possible time. The vertical stability of the plasma with elongation more 1.4 can provided by passive and active stabilization system coils only. The plasma should be approached to passive stabilization coils as near as possible. The simple numerical calculation shows that passive stabilization coils (shown on Fig.1) provide suppression of the vertical MHD instability. Calculations of the discharge scenario show that poloidal flux 6 V.s. stored up into PFC system is enough for Ohmic discharge with $\Delta t = 2$ seconds duration.

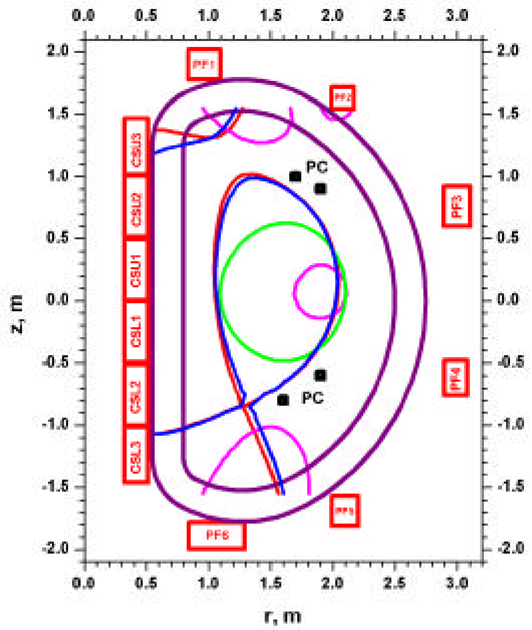

Fig.1

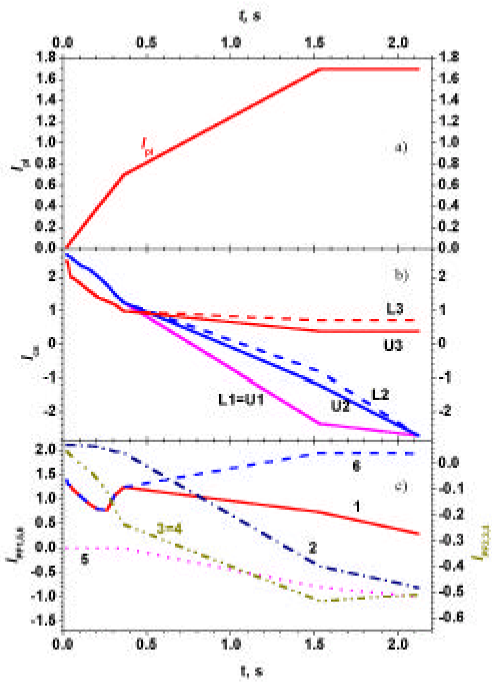

Fig.2

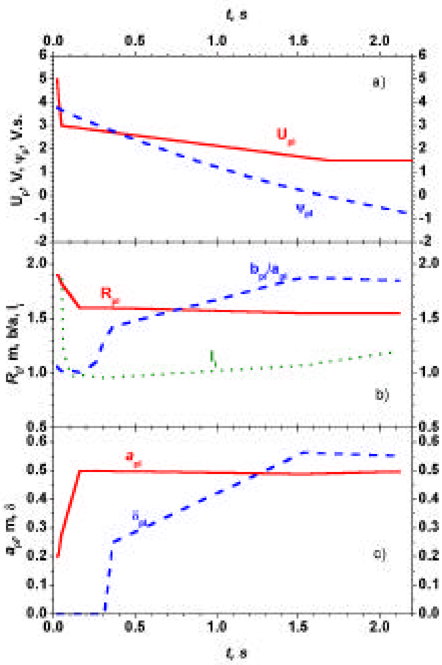

Fig.3