KTM project
(Kazakhstan Tokamak for Material Testing)

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Abstract. The Kazakh Tokamak for Material studies (KTM) is designed for modeling plasma-material interactions in divertor region under conditions expected for ITER. KTM is a tokamak with low aspect ratio A=2. The device enables divertor plates to be changed without disturbance of vacuum. Optimization of power supply sources of equilibrium coils and inductor has been done. The influence of various plasma parameters (density, temperature, effective charge) and final shape of plasma configuration on volt-second of inductor has been studied. Plasma equilibrium has been analyzed with respect to vertical stability. The characteristic times of passive stabilization were obtained and "active" coils were selected for active vertical stabilization. Recommendations were given on passive stabilization coils. A system of position and shape control has been designed.

1. The main goals of KTM and its place in the world fusion program

The main goals of KTM are

- Development of science and technology in Kazakhstan
- Support of Kazakhstan participation as Russia's partner in ITER project.
- Realization of experimental investigations and testing of first wall and divertor materials; studies of different types of particles/heat removal mechanisms under heat fluxes 0.1 – 20 MW/m^2.

One of the main peculiarities of KTM is operative access into the vacuum chamber and possibility of changing the divertor plates without vacuum violation. As a result KTM can be the first fully technological tokamak for fusion materials investigations and testing.

The key research tasks are:

- studies of characteristics of different materials for first wall and divertor;
- studies of sputtering, erosion, arcs, mechanical damages of first wall and divertor plates during both normal operation and disruptions;
- plasma – wall, antenna - plasma – wall interaction;
- various designs of divertor, methods of control fluxes to the divertor, SOL and first wall;
- active SOL formation and its influence on plasma core;
- plasma confinement at aspect ratio A=2 and high ICR heating;
- creation and control of the various plasma shape, optimization of plasma scenarios.
- parameters of plasma core edge and divertor;

Coordination of the research programs of Russian Globus-M, T-15M, TSP-AST tokamaks and KTM gives unique possibilities for support of the ITER project, developing fusion physics and technology.

Creation and putting into operation the KTM tokamak before realization of ITER project can make KTM the basis of international cooperation in plasma material interaction investigations

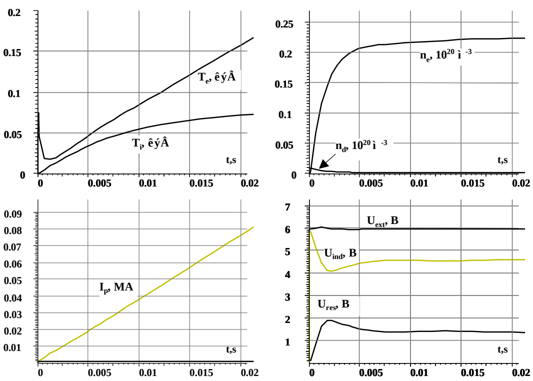
KTM is unique as a Mega-Ampere tokamak with A=2. It can give an important investment to the elongated plasma confinement database for intermediate region A = 2.0÷2.2.

The KTM Parameters and Discharge scenario

Major plasma radius R, m	0.86
Minor plasma radius a, m	0.43
Aspect ratio A	2
Plasma elongation K_{95}	1.7
Toroidal magnetic field at plasma axis $\hat{A}_{to}$, T	1
Plasma current I_P, MA	0.75
Pulse duration Δt_{pl}, s	2-4
Additional heating power P_{aux}, MW	5
Heat flux at divertor plates, MW/m^2	2-20

2. The breakdown and plasma current ramp-up stage

The breakdown region is located near the internal wall ($B_t \sim R^{-1}$; $E \sim R^{-1}$) $R \approx 60\div65$ cm; $a \approx 20$ cm; $k \approx 1$; $Z \approx 0$; $U_{bd} \approx 6\div8$ V; $B_{str,max} < 5\div10$ Gauss ; ($B_{str}/B_t < 10^{-3}$).



To evaluate the ECR power necessary for overcoming ionization and radiation barriers it is necessary to estimate some factors such as: the level of impurities (carbon) in plasma near the end of breakdown $\gamma_{Ñ} \approx 2 \div 4$ % taking into account nonstationary impurity radiation; the ratio of vacuum chamber volume V_v to plasma volume V_o near the end of breakdown is high $V_v/V_o \geq 10$, so that the effect of the cold surrounding gas influx should be evaluated. The SCENPLINT code was used for estimation of ECRH power.

Fig. 1 Sample of temperatures T_e, T_i, densities n_o, n_e, plasma current I_P, voltages U_{ext}, U_{ind} and U_{res} versus time for the initial stage of KTM discharge. ð $=1.8\cdot10^{-3}$ Pa, P_{ECRH} = 70 kW.

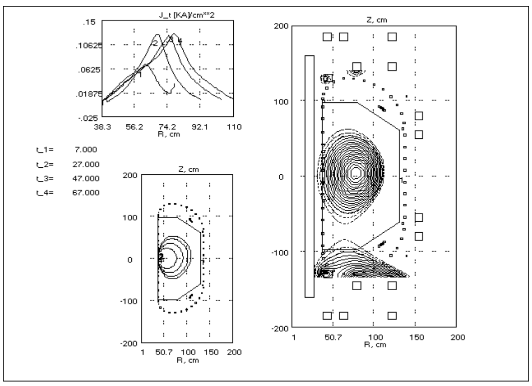


Fig.2 Plasma current ramp-up stage

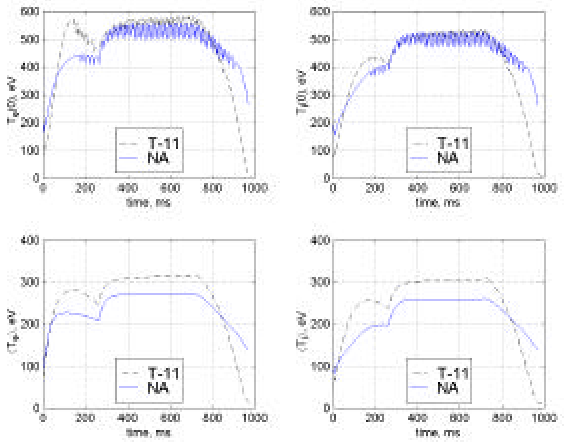
Plasma current ramp-up rate is about $dI_P/dt \approx 3$ MA/s , time duration is 0.25 s. During this stage the plasma cross-section and elongation are increased; plasma moves to the center of the chamber. R = 0.63 0.86 m; a = 0.2 0.43 m; k = 1 1.7. Plasma axis moves in Z direction. Z_{axis} = 0 0.13 m. During current ramp-up the safety factor is $q_a \approx 10\div5$. The DINA code was used for modeling this stage. (See description of DINA coder in refs [1-3]).

- To decrease the MHD activity and to prevent from runaway it is important to correlate the plasma density n_e and current I_P: $n_e(t) \approx (0.3 - 0.5)\cdot I_P(t)/\pi a^2$
- At the plasma current plateau the plasma density $n_e = 0.5\cdot10^{20}$ ì$^{-3}$ does not exceed the Murakami and Greenwald limits ($n_M = 2\cdot B_t/q\cdot R = 0.6\cdot10^{20}$ ì$^{-3}$, $n_G = I_P/\pi\cdot Ka^2 = 0.7\cdot10^{20}$ ì$^{-3}$)

3. Plasma parameters for OH heating and for additional heating

The DINA code was used with two variants of energy confinement scaling: Ò-11 scaling and Neoalcator (NA) scaling.

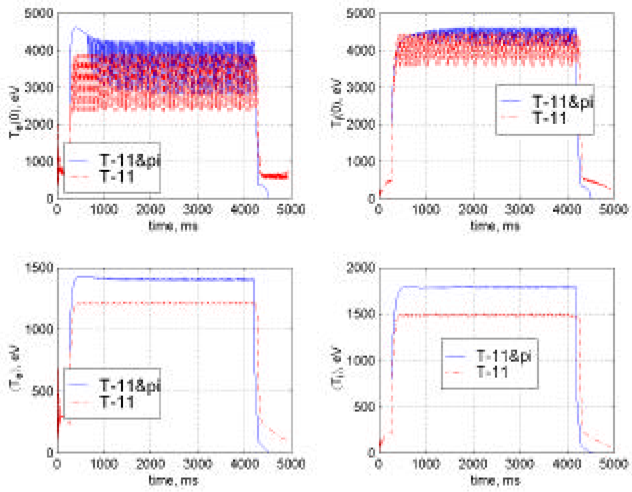
scaling	NA	T-11
l_i	0.84	0.80
β_p	0.20	0.23
β_N	0.67	0.67
τ_E, ms	42	57
P_{OH}, MW	1.0	0.85
U_p, V	1.3	1.1
$\Delta\Psi_{res}$, V·s	0.6	0.5
$\overline{T}_e$, eV	270	315
$T_e(0)$, eV	560	550
W_p, kJ	42	48



In the case of pure OH heating the duration of plasma current plateau does not exceed $\Delta t_{plateau}$= 0.5 s for $\Delta t_{plateau}$= 0, it is possible to reach the plasma current I_P = 1 MA.

- ICRH will be used for plasma heating (f ≈ 13 MHz, $P_{absorbed}$ ≈ 5 MW). For KTM the L-H transition threshold takes place at P_{thr} ≈ 0.6 MW. The DINA code gives the following results: β_N ≈ 3 (value defined by plasma confinement), T_e, T_i ≈ 3 – 4 keV, plasma thermal energy W_P ≈ 0.2 MJ, loop voltage U_P ≈ 0.15 – 0.2 V, pulse duration Δt_{pulse} ≈ 4 s.

scaling	Ò-11
l_i	0.84
β_p	0.85
β_N	2.9
τ_E, ms	34.5
P_{OH},MW	0.13
$\Delta\Psi_{res}$, Vs	0.70
$\overline{T}_e$,eV	1200.
$T_e(0)$,eV	3200.
P_{RF},MW	5.0
W_p,MJ	0.18



- The value of the achieved normalized beta β_N defines to some degree the pulse duration. So, the limits on β_N was the subject of a special analysis.

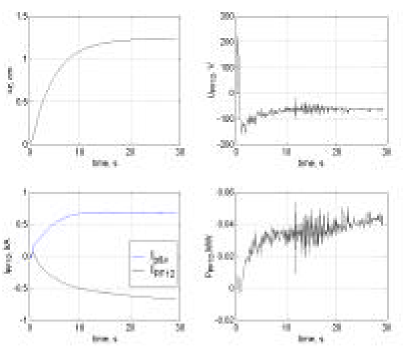
→Theory - <u>absolute limits</u>: $\beta_N \le 8/A^{1/2}$; $\beta_N \le 12/A \Rightarrow \underline{\beta_N \le 5 - 6}$ for À ≈ 2;
→Experiments - <u>database in the whole</u>: $\underline{\beta_N \le 3.5}$;
→Experiments - <u>l_i dependence</u>: $\beta_N \le 4 \cdot l_i \Rightarrow \underline{\beta_N \le 3.2 \div 3.4}$ at l_i = 0.8 – 0.85
→Experiments - <u>q_a dependence</u>: $\beta_N/q_a \le 0.6\text{-}0.8 \Rightarrow \underline{\beta_N \le 2.4 \div 3.2}$ at $q_a \approx 4$
→Experiments - <u>neoclassical islands</u>: $\beta_N \le f(\nu_e^*) \Rightarrow \underline{\beta_N \le 2.4 \div 3}$ at $\nu_e^* \approx 0.02\text{–}0.05$

For KTM β_N ≈ 3 is assumed for P_{aux} ≈ 5 MW. In this case Δt_{pulse} ≈ 4 s (solenoid flux swing ≈ 2 Wb). If the real value of β_N is lower the pulse duration will decrease (for β_N ≈ 2 – down to ≈2 s). If β_N>5÷6 Δt_{pulse} will be limited by the allowable magnetic coils heating.]

4. Divertor heat fluxes

- The heat flux distribution between the divertor and the first wall is assumed to be 70% and 30% respectively. Outboard/Inboard ratio is 4/1.
- For the additional heating P_{aux} = 5 MW, the SOL thickness is about $\Delta_{SOL} \approx 3$ cm, flux expansion ≈ 3, distribution of power is 1.5 MW (wall), 2.8 MW (outboard plate), 0.7 MW (inboard plate). The heat flux on the outboard plate - $P_{div} \approx 2.8\ \mathrm{MW}/(2\pi R\Delta_{SOL} f_{flux}) \approx 6\ \mathrm{MW/m^2}$

5. Plasma vertical position control



The natural plasma elongation K_{nat} for aspect ratio À = 2 is about 1.2÷1.3 in KTM. The reference value of plasma elongation in KTM is K_{95} = 1.7. So, plasma position control in vertical direction is necessary. The resistance of the vacuum chamber ($R_{vvt} \approx 120\ \mu\Omega$) gives vertical instability time $\tau_\gamma \approx 1.5 - 2$ ms (increment $\gamma \sim 500 - 600\ s^{-1}$). Passive conducting rings inside vacuum volume lead to $\tau_\gamma \approx 3$ ms ($\gamma \sim 300\ s^{-1}$). In this case active plasma control can be used. The DINA code was used to estimate parameters of the active control system.

Fig. 3 Modeling of passive and active control of vertical position.
(HFC coils for plasma shape control is showed as PF12.)

6. Design

The KTM tokamak consists of 20 toroidal field coils (TF), the vacuum vessel (VV), the central solenoid (CS), 18 poloidal field coil (PF), the divertor, the vacuum sluice (VS) and the support.

There are 20 one-turn D-shaped TF-coils. Each coil consists of four parts and has four joints. The presence of upper and bottom joints allows a fully-welded vacuum vessel, facilitates assembly-disassembly and also allows one to install the poloidal coils between the vacuum vessel and the toroidal field coils. Bronze is used for the inner leg. The other parts of the TF coil are manufactured from copper. The inner legs of TF coils form the central column. The intercoil structures are placed between the TF coils to ensure normal operation under electromagnetic loads. They form bearing rings located from above and from below and on outer contour of a machine.

Calculations of electromagnetic loads on the TF coil have been carried out. The distribution of unit length force acting on the center line of the TF coil has been obtained. 2D FE models have been used for analysis of the connections between parts of the TF coil and other structures. The water cooling is used. The analysis of the calculations shows that all rated stresses do not exceed the limit stresses. The main parameters of the vacuum vessel: inner radius – 0.38 m, outer radius – 1.45m, height – 2.6 m. The disposition of ribs in combination with a thin shell provides the necessary parameters of the vacuum vessel - 115 $\mu\Omega$ and margin ~5 for stability. 3D FE models have been used for structural analysis of the vacuum vessel. The calculations show that the selected dimensions of the vacuum vessel elements obey the static strengths and the effective stresses in structural elements do not exceed the allowable values. The thermo-mechanical calculations of a steady-state temperature status of the vacuum vessel shell, of graphite tiles protecting the inner part of the vacuum vessel from plasma heat flux and of in-vessel components have been carried out.

The calculations show that TF inner legs are needed for cooling and it is recommended to cool the vacuum vessel wall near the divertor. The vacuum vessel is equipped with the baking system. The heat insulation consists of separate segments, which can be dismantled easily during operation.

The divertor is intended for controlling plasma contamination and receiving particles and energy flux from the main plasma. The divertor consists of removable plates and table which can move and rotate. The replacement of the removable plates is made without violation of vacuum. The removable plates are placed in the loading-unloading zone with the help of vertical motion and rotation of the table in horizontal plane.

The central solenoid (CS) is located inside of 560 mm diameter bore of the central column. The central solenoid consists of 512 turns winded by four layers. The conductor from CuAg alloy of HKO1 type manufactured by OUTOKUMPU OY in Pori, Finland is selected (yield strength 270 MPa, ultimate tensile strength 330 MPa) as a material for winding of the central solenoid. The conductor has a central hole for water cooling. Five pairs of poloidal field coils (PF1-PF5) and the central solenoid are used to obtain various scenarios of variation of the plasma loop voltage and the necessary plasma configuration. Two additional coils (HFC) are used for plasma shape control. For breakdown and plasma shaping three pairs of poloidal field coils (CC1-CC3) are used to compensate the central solenoid stray field. The poloidal field coils are located between the vacuum vessel and the toroidal field coils (TF). The material for the poloidal field coils is copper bus with a central hole for water cooling.

7. Conclusion

The KTM tokamak modeling and design have shown the physical validity of the project, high technical and technological development of the machine units such as the vacuum vessel with in-vessel components, the vacuum sluice and the divertor, the central solenoid and the central column of the toroidal field coil. Studies of protective properties of modern and perspective materials is one of the main tasks of the machine.

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