Control of the Plasma Fluxes into the Divertor Region of the Tokamak KTM

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Abstract. The Kazakhstan Tokamak for Material studies (KTM) being under construction now is dedicated for extensive investigations of the plasma facing components in the conditions of ITER and future fusion reactors. The problem of the control of the plasma fluxes into the divertor region and onto the target plates is especially important for this device. In the present paper, various factors determining the structure and the values of such fluxes are analyzed. The main factors are the additional heating power and the plasma current. The additional heating power determines the SOL width and the SOL broadening at the divertor plates is determined by the plasma current. Using the ITER scalings and the numerical code DINA it is shown that the peak power at the divertor plates can change in a wide region including the peak loads of ITER. The influence of the x-point sweeping, the vertical displacement of the divertor table, ELMs and VDE are also considered.

1. Introduction.

Low aspect ratio tokamaks [1] enables us to obtain high parameters of the hot plasma in small devices and seems to be an alternative way to the nuclear reactor [2] or volumetric neutron source [3]. The spherical tokamak can also be attractive from another point of view. It is well known that the choice of plasma facing materials is one of the main problems in developing of a tokamak reactor. This problem are under investigation on the existing large tokamaks. However, since these devices are not suitable for such studies the volume of the results obtained does not match the importance of the problem.

The analysis has shown that the simplest and cheapest way to solve the problem is to create a special material oriented tokamak with a low aspect ratio [4]. Such specialization leads to some peculiarities in its design. In the Kazakhstan Tokamak for Material studies (KTM) being under construction now, an additional space is provided for a special divertor device which enables us to replace the divertor plates without opening the vacuum chamber (*FIG. 1*). It will be done by moving the divertor table (*FIG. 2*) up to the tokamak middle plane where the samples can be get in and out with the help of a special vacuum sluice. All this enables us to substantially increase the body of material studies. To solve material tasks it is reasonable to increase the aspect ratio A up to 2. It will enable us to model the ITER conditions and at the same time KTM will become durable and reliable device which is very important for material studies.

2. Design and plasma parameters of KTM.

Some “thrifty” construction of installation has been provided in order to meet these requirements [5]. Low aspect ratio allows small dimensions of the installation i.e. minor radius a=0.43 m, major radius R=0.86 m, vessel vertical elongation of ~3, plasma column cross-section elongation k=1.7. Single-null magnetic field configuration with null at vessel underbody area is accepted in KTM, as well as in ITER. All metal surfaces facing hot plasma are planned to be shielded by carbon-graphite coating.

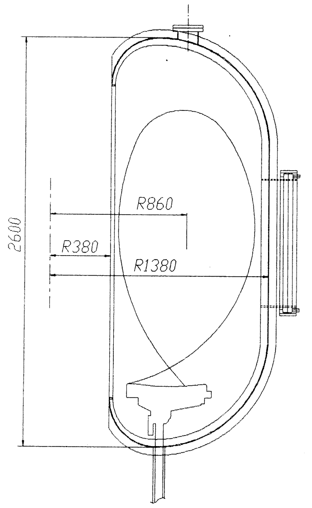

FIG. 1. KTM tokamak cross section.

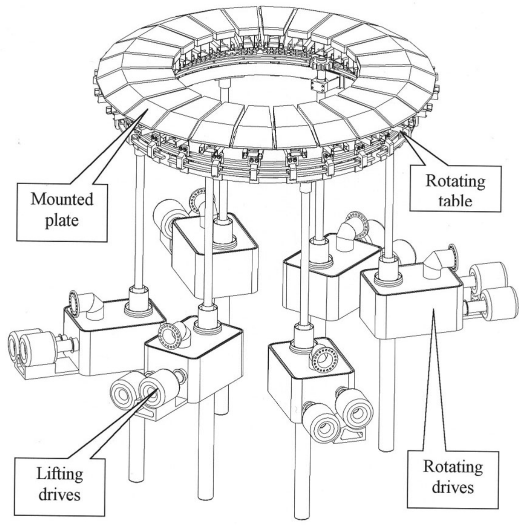

Fig. 2. Divertor table of KTM.

Since KTM aspect ratio is not too low, this would allow inductor construction to be less tensed and to reserve more volt-seconds (2.5 V·s). Taking this reserve of magnetic flux the installation is capable of operating in different modes. Ohmic mode has ~1 MA and prompt pulse of ~ 1 s. It is possible to reduce the current down to 0.75 MA, but increase pulse duration up to 4-5 s This may be attained via RF ion heating (P_{AUX}~5 MW) and, finally, the current can be reduced down to 0.3 MA while pulse duration increased to 10 s (P_{AUX} ~5-7 MW). Low aspect ratio allows using fairly low toroidal magnetic field B_t not eliding to plasma stability. For B_t=1 T and I_p=0.75 MA safety factor q makes ~6. This value of B_t will require rather acceptable level of installation energetics. Electron and ion temperature in plasma column is determined by additional heating power and its value depends on energy lifetime (i.e. scaling). T-11 scaling calculations subject to 5 MW additional heating provide the maximum electron and ion temperatures of up to 5 keV and 1.5-3 keV respectively. Expected plasma column parameters at I_p=0.75 MA are cited in *Table I.*

Table I. Basic Parameters of the KTM tokamak.

Plasma major radius R	0.86 m
Plasma minor radius a	0.43 m
Aspect ratio A	2
Plasma elongation k_{95}	1.7
Toroidal magnetic field B_t	1T
Plasma current I_p	0.75MA
Additional RF-heating power P_{AUX}	6-7 MW
Pulse duration t_p	2-4 s
Plasma density n	$(3\text{-}5)10^{19}m^{-3}$
Plasma temperature T_0	(1.5-3) keV
Plasma safety factor q_{95}	4-6
Triangularity δ	0.3-0.5

Poloidal coils parameters (current and position) are selected so that to form divertor configuration of magnetic field (*FIG. 1*), ensure maximum plasma filling in vacuum vessel, provide plasma column stability to vertical shifts, and to obtain possibility to control plasma flux on divertor plates. Separatrix closeness to vessel walls was determined so that convective heat flux from plasma column would not reach the upper part of the vessel. Here it is necessary to place passive stabilization coils (130 $\mu\Omega$ resistance) near the plasma column in order to ensure vertical stability. This allows column stabilization via feedback system.

3. Plasma Fluxes into the Divertor.

It is also necessary to control the plasma fluxes into the divertor region. It will extend the potential of material studies and improve their scientific level. The power and the structure of the plasma fluxes can be controlled by various means:

- varying the value of additional ICR heating;
- varying the plasma current;
- swiping the x-point;
- moving the divertor in the vertical direction;
- initiating plasma disruptions;
- varying the properties of ELMs.

The analysis of these possibilities is presented in this paper. The structure of the plasma flux into the divertor region is known to be determined mainly by the SOL width, the pitch angle and the magnetic tube broadening near the target, f_{exp}. The theoretical description of these values is not fully adequate to the experiment. For this reason "tokamak" community has developed a scalings to determine SOL thickness [6] by analogy with the scaling determining energy confinement time. SOL thickness scalings represent phenomenological expressions for calculation of SOL thickness in different tokamak operation modes i.e. L and H modes. λ_q in L mode can be expressed differently depending on power delivered to divertor or power of additional heating delivered to tokamak. For the first case we have:

$$\lambda_q^{L-1}(m) = (6.6 \pm 2.2)10^{-4} R(m)^{1.21\pm0.15} P(MW)_{div}^{-0.19\pm0.05} q_{95}^{0.59\pm0.11} \overline{n}_e (10^{19} m^{-3})^{0.54\pm0.15} Z_{eff,scal}^{0.61\pm0.09} \quad (1)$$

for the second variant:

$$\lambda_q^{L-2}(m) = (7.2 \pm 2.4)10^{-4} R(m)^{1.21\pm0.15} P(MW)_{TOT}^{-0.28\pm0.08} q_{95}^{0.59\pm0.11} \overline{n}_e (10^{19} m^{-3})^{0.68\pm0.16} Z_{eff,scal}^{0.65\pm0.09} \quad (2)$$

There are analogous expressions for H mode:

$$\lambda_q^{H-1}(m) = (5.2 \pm 1.3)10^{-3} P(MW)_{div}^{0.44\pm0.04} B(T)^{-0.45\pm0.07} q_{95}^{0.57\pm0.16} \quad (3)$$

$$\lambda_q^{H-2}(m) = (5.3 \pm 1.4)10^{-3} P(MW)_{TOT}^{0.38\pm0.04} B(T)^{-0.71\pm0.08} q_{95}^{0.30\pm0.15} \quad (4)$$

All expressions were obtained by analysis of experimental data from DIII-D, JT-60 and ASDEX-Up. The publication on ITER [7] contained different expressions for $\lambda_{q,}$ for example:

$$\lambda_q^H(m) = 1.82 \cdot 10^{-3} P_{div}^{0.35}(MW) q_{95}^{0.77} R^{0.46}(m) \quad , \quad (4a)$$

For ITER conditions they provide values close to expressions (1-4), however, KTM conditions break such agreement. Therefore it could be expedient to cite these new expressions and perform comparative calculations.

Prior to SOL thickness estimation using these formulas one should determine the ratio of energy flux power values delivered at outer and inner strike-points. Generalization of experimental data obtained at major installations demonstrates that the average flux onto external part of divertor plate is 2-4 times as high compared to that onto internal part. At the same time spherical tokamaks exhibit the tendency to having a larger ratio. For instance, at START tokamak this ratio is ~5.5 while the expected value is even higher [2]. To be definite, let us assume that heat flux in external strike-point suburb makes ~80% of the flux to divertor.

Because additional heating power in KTM was provided at P_{AUX}=5MW the external part of divertor plate shall receive 2.8 MW. For typical KTM parameters while operating in L-mode (1) provides λ_q=0.5 cm. For H-mode (3) gives λ_q=2.4 cm. According to these estimations SOL thickness in H mode is larger than that in L-mode. However it should be noted that in KTM L-H transition threshold is fairly low. L-H transitions scalings have been also developed. As it was mentioned above the expert group for ITER-EDA proposed the following expression [8]:

$$P_{L-H}(MW) = 0.4\bar{n}_e(10^{20}\,m^{-3})B_t(T)R^{2.5}(m) \qquad (5)$$

For KTM scaling (5) provides $P_{L\text{-}H}$~0.13 MW. According to data cited in [9] being in good agreement with JET experiments [10] one may write the following expression:

$$P_{L-H}(MW) = 0.76\bar{n}_e^{0.75}(10^{20}\,m^{-3})B_t(T)R^2(m)A^{-1}(a.u) \qquad (6)$$

where A is hydrogen isotope mass in atomic units. The latter scaling gives $P_{L\text{-}H}$~0.3 MW (A=1) for KTM parameters.
Recently improved scaling for L-H transition is introduced in [11] as the following:

$$P_{L-H}(MW) = \left[(0.9\pm0.2)\times0.6^{\alpha}\right]A^{-1}(a.u.)B_t(T)\bar{n}_e^{0.75}(10^{20}\,m^{-3})R^{2.}(m)(\bar{n}_eR^2)^{\alpha} \qquad (7)$$

where –0.25≤α≤0.25 ensures the maximum value of L-H transition threshold in KTM (α=-0.25) ~0.62 MW. Therefore in KTM normal operation conditions at 5 MW additional heating tokamak “should” operate in H mode. In case power of additional heating is less than 0.6 MW KTM should operate in L mode and thus SOL thickness for L mode must be recalculated with an allowance for this circumstance: the thickness increases up to 0.7 cm.

Heat and particle flux density values depend primarily on magnetic field structure near divertor plates as well as on the angle between magnetic lines and plate surface. Due to geometrical expansion of space between magnetic surfaces the thickness of boundary plasma and, therefore, the width of energy flux to divertor plate increases. Numerical calculations of magnetic field structure for different q values and, thus, for different plasma currents have demonstrated degradation of f_{exp} value with growth of q (see *Table II*).

Table II.

	I	II	III
I_P, MA	0.75	0.6	0.5
q_{edge}	6.3	8.2	10.3
f_{exp}	5-6	4-5	3-4
$\frac{\lambda_q^{H-1}}{W}, \frac{cm}{MW/m^2}$	$\frac{1.9}{5.3}$	$\frac{2.2}{5.8}$	$\frac{2.4}{7}$
$\frac{\lambda_q^{H}}{W}, \frac{cm}{MW/m^2}$	$\frac{0.97}{10.4}$	$\frac{1.2}{10.6}$	$\frac{1.4}{12.1}$

Minimum f_{exp} is approximately 3 (I_p=0.5 MA). The average width of thermal irradiation zone on divertor Δ shall be approximately 7.9 cm while average heat flux intensity shall make W=2.8 MW /(2πRΔ)=7 MW/m^2 (see *Table II*). In case plasma current is high (0.75 MA) q~6.3, f_{exp} increases up to 5 while the intensity falls down to ~5.3 MW/m^2. These values may be lower at smaller angle between magnetic field and divertor plate’s surface. In *Table II* this limiting intensities are compared with intensities, obtained with (4a). So, from (1-7) and *Table II* it follows, that the power density value monotone increases with P_{AUX} and decreases with plasma current. Heat loads, cited above, were calculated under assumption that heat fluxes to divertor volume are time-constant during discharge. It is well known that H mode (KTM design mode) is usually accompanied by ELM. Such instabilities result in heat flux modulation. In case off-duty ratio of this process $((\tau\nu)^{-1}$, where τ is ELM duration and ν is the frequency) is high it results in short but drastic rise of heat flux intensity to divertor plates. This instability is represented by various modes, which mainly differ in frequency and modulation depth. Frequency $10 \leq \nu \leq 100$ Hz decreases with higher triangularity of plasma column and τ duration makes approximately 100 μs. Energy of low frequency ELM ($\nu \sim 10$ Hz) contains 3-7% of energy stored in plasma column. At the same time peak heat flux at divertor reaches 150-300 MW/m^2. Higher frequency ELMs transfer less energy and are considered safe in respect to ITER. So far it is necessary to estimate the value of most dangerous heat loads emerging at disruption. Assuming that instability duration is ~100 μs and KTM energy store makes ~0.15MJ one would find $W=3 \div 6$ GW/m^2.

The structure of the plasma flux onto the divertor plate can be changed by moving the x-point or the divertor table. In the first case, the numerical calculations show that the KTM poloidal coils enable us to move the strike points by 3-5 cm without changes of main discharge parameters. But magnetic field structure changes substantially near divertor plates. When X-point is displaced to torus axis the flux expansion f_{exp} decreases, therefore, power density increases. When X-point is displaced away from torus axis the flux expansion greatly increases and, thus, power density decreases. In the second case, the SOL width increases with the elevation of the divertor table and the maximum power density region moves smoothly to the axis of the torus. The value of the maximum power density decreases by 3-4 times in this case.

Conclusion.

1. Selected aspect ratio (A=2) and plasma column cross-section elongation (k=1.7) enable us to use central solenoid with acceptable technical specifications, which provides 0.75 MA input current and its maintenance during 4-5 s.
2. Divertor unit design allows changing plasma-wall interaction conditions in a wide range of parameters and enables extension of material study database being of interest of ITER and further tokamak-reactors.

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