STUDIES OF THE DISRUPTION PREVENTION BY ECRH AT PLASMA CURRENT RISE STAGE IN LIMITER DISCHARGES

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Abstract

Studies of disruption prevention by means of ECRH in T-10 at the plasma current rise phase in limiter discharges with circular plasma cross-section were performed. Reliable disruption prevention by ECRH at HF power $(P_{HF})_{min}$ level equal to 20% of ohmic heating power P_{OH} was demonstrated. m/n=2/1 mode MHD-activity developed before disruption (with characteristic time ~120 ms) can be considered as disruption precursor and can be used in a feedback system.

1. INTRODUCTION

The influence of ECRH on the behavior of plasma MHD-activity on T-10 was investigated at the phase of plasma current I_p rise aimed the reliable prevention of disruption. The goals of experiments were as follows:

(1) Demonstration of reliable disruption prevention by ECRH. Determination of minimum HF power launched into plasma required for reliable disruption prevention.
(2) Study of disruption precursors which can be used in a feedback system for disruption prevention.
(3) Determination of HF power switch-on moment and HF pulse duration required for the reliable disruption prevention.

The experiments performed are the continuation of previous studies on MHD-activity stabilization by means of ECRH [1,2].

In the previous T-10 experiments (circular plasma cross-section, limiter discharges) it was shown [3] that at the given toroidal field B_T and plasma current growth rate $\partial I_p/\partial t$ there exists the maximum plasma density growth rate $(\partial n/\partial t)_{lim}$ (maximum influx from the gas puffing valve). If the $\partial n/\partial t$ value exceeds the maximum one, the rapid monotonous growth of m/n=2/1 MHD-mode is observed with the consequent development of disruption instability.

Such MHD-activity behavior allows us to suggest that at $\partial n/\partial t>(\partial n/\partial t)_{lim}$ a disruption instability is developed because of cooling of plasma periphery by gas influx from the valve Γ_v, shrinkage of plasma current profile and going out of the q=2 surface onto the plasma boundary.

2. THE EXPERIMENTAL CONDITIONS

At the first stage of experiments at the given current growth rate $\partial I_p/\partial t=0.60\div0.65$ MA/s and $B_T=2.4$ T (on-axis ECRH) the value of $\partial n/\partial t=6\cdot10^{19}\ m^{-3}\cdot s^{-1} < (\partial n/\partial t)_{lim}=10\cdot10^{19}\ m^{-3}\cdot s^{-1}$ was chosen so that it provides the stable discharge development (these $\partial n/\partial t$ values correspond to influx levels from gas puffing valve $\Gamma_v=1.6\cdot10^{20}\ s^{-1}$ and $2.7\cdot10^{20}\ s^{-1}$, respectively). Then the auxiliary gas puffing was switched-on at $t_{g.p.}$=200 ms instant which provided the rapid increase of the density growth rate $\partial n/\partial t > (\partial n/\partial t)_{lim}$.

"Regular", i.e. occured at the same time, disruption in ohmic discharge was obtained at $\partial n/\partial t = 12.3 \cdot 10^{19}$ $m^{-3} \cdot s^{-1} \cong 1.25(\partial n/\partial t)_{lim}$ ($\Gamma_v = 3.3 \cdot 10^{20}$ s^{-1}). The valve was open until the end of current rise phase so simulating prolonged non-controlled impact of gas and impurity flows from chamber walls causing the disruption.

The MHD-activity behavior was investigated for HF pulse switch-on at different instants t_{HF}, at different P_{HF} power levels and at a different localization of HF power deposition.

Because the character of a real non-controlled impact on plasma causing the disruption is not exactly known, during the second stage of experiments the disruption prevention was studied at auxiliary gas puffing valve switching-on for a short time (50 ms) so simulating in this way a brief non-controlled impact on plasma. In this case regular disruption in ohmic discharges was obtained at the value $\partial n/\partial t = 22 \cdot 10^{19}$ $m^{-3} \cdot s^{-1} \cong 2.5(\partial n/\partial t)_{lim}$ ($\Gamma_v \cong 5.9 \cdot 10^{20}$ s^{-1}).

3. THE RESULTS OF EXPERIMENTS WITH PROLONGED GAS PUFFING

3.1 The results of experiments on determination of minimum value of HF power sufficient for disruption prevention are presented in fig.1. In these experiments HF pulse was switched-on simultaneously with auxiliary gas puffing valve ($t_{g.p.}$=200 ms). The main results of these experiments are as follows:

(1) At auxiliary gas puffing valve switching-on in OH regime with $\partial n/\partial t = 1.25(\partial n/\partial t)_{lim}$ the monotonous growth of m=2 MHD-mode was observed during 120 ms; after which thermal quench occured, i.e. fast and large decrease of temperature and density (with afterwards current disruption). Characteristic time of m=2 island development decreased to 40-50 ms with shift of valve switching-on instant to the start-up of the discharge.

(2) HF pulse switching-on results in reliable prevention of disruption if HF power value P_{HF} exceeds the level $(P_{HF})_{min} \approx 80$ kW which constitute ~20% of ohmic heating power P_{OH} (in current plateau stage) (fig.2a).

As shown in fig.1b HF pulse even with high value of $P_{HF} \approx (6 \div 8)(P_{HF})_{min}$ decreases the rate of m=2 mode development but does not prevent process of plasma periphery cooling. That's why $(P_{HF})_{min}$ value required for disruption prevention depends on influx duration from auxiliary gas puffing valve (i.e. on duration of non-controlled impact on plasma). And that's why shift of valve switching-on instant toward the discharge start-up ($t_{g.p.}$=100 ms) causes the $(P_{HF})_{min}$ value increase as shown in fig.2c.

(3) The maximum level of m=2 MHD-mode amplitude A_{mhd}^{lim} at which the disruption occurs changes weakly despite a significant changing of conditions causing the disruption. As was shown previously during studies of limit density on T-10 [3] this maximal level A_{mhd}^{lim} corresponds to the condition when m=2 island width is close (consists 70-80%) to the distance between magnetic surface q=2 and plasma boundary.

(4) The reliable disruption prevention at $P_{HF} = 1.2(P_{HF})_{min}$ takes place at HF pulse switching-on until the instant when the amplitude of MHD-mode reaches the level not smaller than 70% of limit one. More earlier switching-on the HF pulse does not result in significant changing of $(P_{HF})_{min}$.

3.2 The goal of experiments with changing of B_T, i.e. the displacement of HF power deposition region in plasma, was to try to decrease the value of HF power sufficient for disruption prevention $(P_{HF})_{min}$ by means of plasma heating in m=2 island neighborhood. But as shown in fig.2b $(P_{HF})_{min}$ value increases weakly with displacement of HF power deposition region from plasma center up to $\Delta R \cong 20$

cm (which is close to q=2 surface). Further displacement causes abrupt increase of $(P_{HF})_{min}$ value that can be associated with decreasing of HF power absorbed in one pass (in accordance with calculations). The part of HF power not absorbed in plasma can result in the increase of additional gas and impurity influx from chamber wall and other components.

4. PECULARITIES OF EXPERIMENTS WITH A SHORT GAS PUFFING PULSE

As was mentioned above in section 2 in case of a short gas pulse (50 ms) the “regular” disruption in OH regime was achieved when gas influx rate exceeded a limit value $(\partial n/\partial t)_{lim}$ by a factor of ~2.3. Naturally, such a strong increase of impact on plasma (2.3 instead of 1.25) causes significant increase of minimal HF power value required for disruption prevention $(P_{HF})_{min}$, as is seen in fig.2d. In this case the shift of gas puffing valve switch-on instant $t_{g.p.}$ does not cause significant changes in $(P_{HF})_{min}$ value (in contrast to the experiments with prolonged impact on plasma).

5. COMMENTS

Overall picture of described above results corresponds to the view that the HF power deposited in plasma weakens (slowing down) a process of plasma periphery cooling and current channel shrinkage which results in reliable disruption prevention.

The results obtained allows us to consider the m=2 MHD-mode amplitude (with characteristic time ~100 ms) as a disruption precursor and to use it in a feedback system for disruption prevention. The correlation between the limit level of MHD mode amplitude A_{mhd}^{lim} and ratio of the m=2 island width and distance from q=2 surface to plasma boundary allows us to calculate A_{mhd}^{lim} value what is necessary for a feedback system design.

The disruption in tokamaks can arise also at β-limit achievement or as a result of plasma column vertical stability loss. Feasibility of such disruption prevention (i.e. extending stability region) by means of ECRH was not considered in this work and requires a special experimental investigation.

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FIGURE CAPTIONS

Fig.1 Traces of plasma density $n_e(t)$, soft X-ray intensity $I_{SXR}(0,t)$, m=2 MHD-mode amplitude A_{mhd} at different HF power P_{HF} levels (t_{dis} is the instant of disruption).

Fig.2 Dependence of m=2 mode amplitude on HF power P_{HF} (a), changing of $(P_{HF})_{min}$ as a function of toroidal field B_T (b), valve switching-on instant $t_{g.p.}$ (c) and $(\P n_e/\P t)/(\P n_e/\P t)_{lim}$ parameter (d). In fig.2a A_{mhd} values are given at different instants: $(t_{dis})^{OH}$ – disruption instant in OH discharges; $(t_{dis})^{ECH}$ – disruption instant in ECRH discharges; t_{max} – $(A_{mhd})^{max}$ instant in discharges without disruption

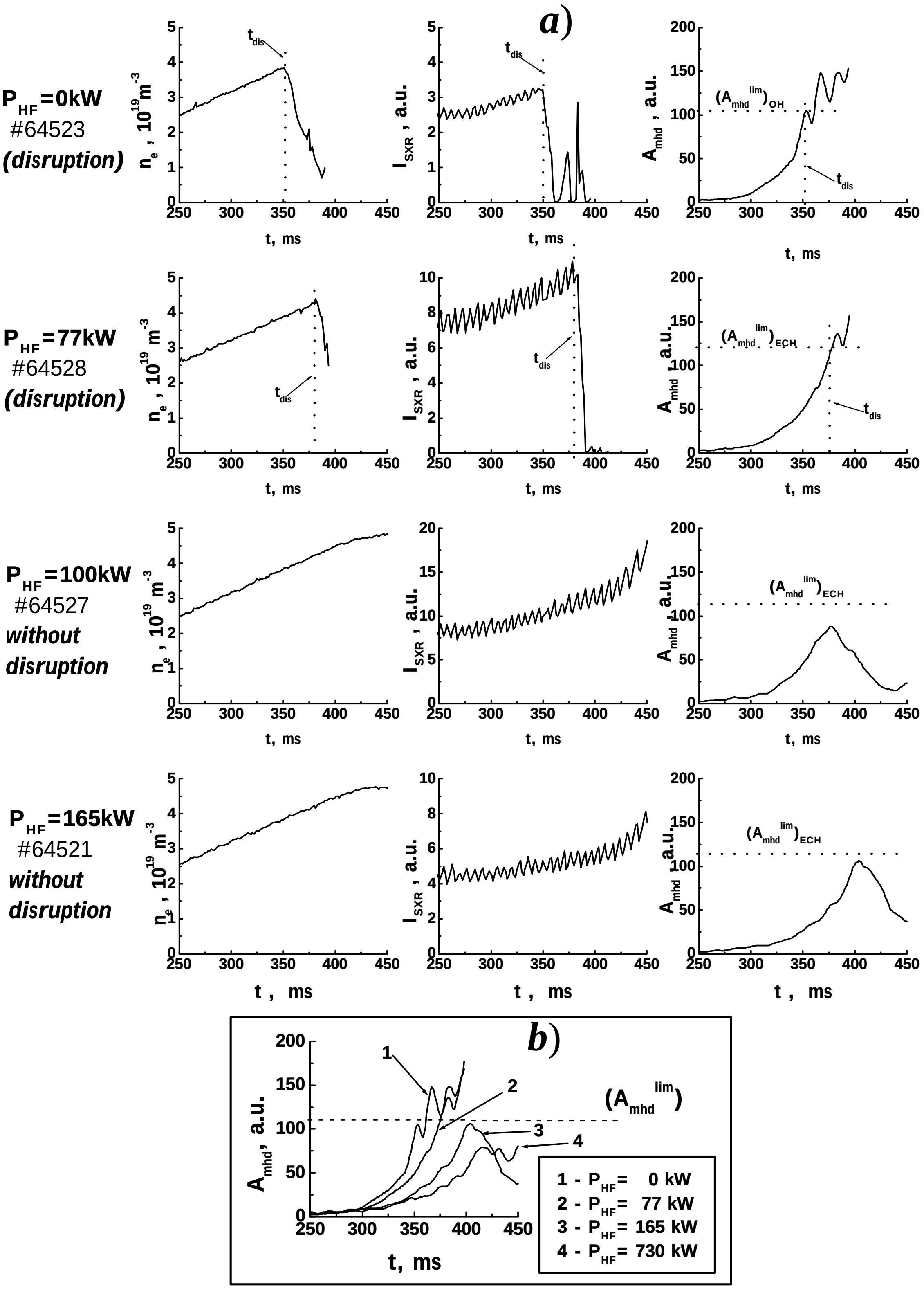
a)
P_HF=0kW
#64523
(disruption)
P_HF=77kW
#64528
(disruption)
P_HF=100kW
#64527
without disruption
P_HF=165kW
#64521
without disruption
n_e , 10^19 m^-3
I_SXR , a.u.
A_mhd , a.u.
t, ms
t_dis
(A_mhd^lim)_OH
(A_mhd^lim)_ECH
0
50
100
150
200
250
300
350
400
450
b)
(A_mhd^lim)
1
2
3
4
1 - P_HF= 0 kW
2 - P_HF= 77 kW
3 - P_HF= 165 kW
4 - P_HF= 730 kW

Fig.1

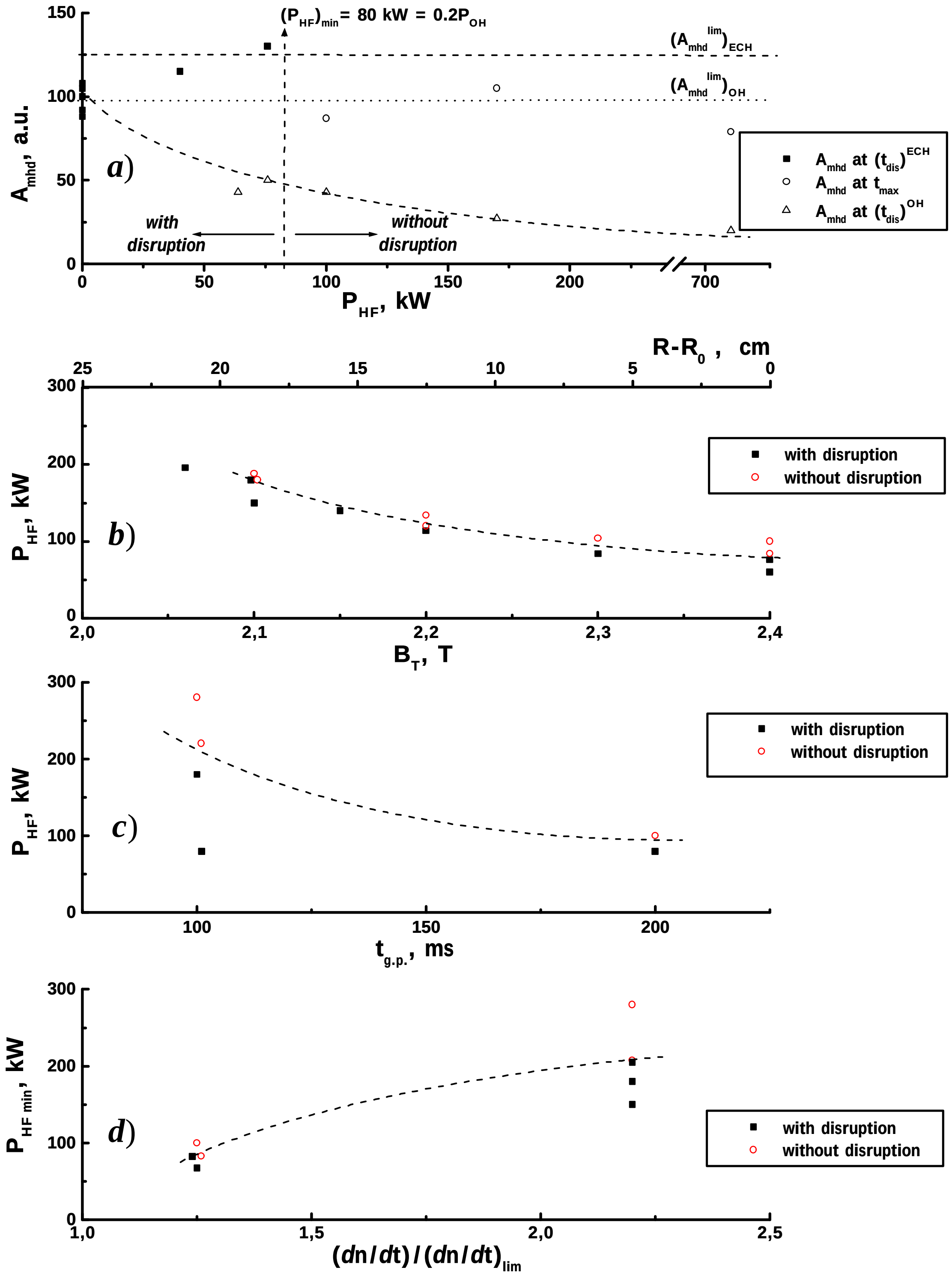
(P_HF)_min = 80 kW = 0.2P_OH
(A_mhd^lim)_ECH
(A_mhd^lim)_OH
A_mhd, a.u.
a)
with disruption
without disruption
A_mhd at (t_dis)^ECH
A_mhd at t_max
A_mhd at (t_dis)^OH
P_HF, kW
0
50
100
150
200
700
R-R_0 , cm
25
20
15
10
5
0
P_HF, kW
b)
with disruption
without disruption
B_T, T
2,0
2,1
2,2
2,3
2,4
P_HF, kW
c)
with disruption
without disruption
t_g.p., ms
100
150
200
P_HF min, kW
d)
with disruption
without disruption
(dn/dt)/(dn/dt)_lim
1,0
1,5
2,0
2,5
300
200
100
0

Fig.2

POSSIBILITY OF AN INTERNAL TRANSPORT BARRIER PRODUCING UNDER DOMINATING ELECTRON TRANSPORT IN THE T-10 TOKAMAK

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Abstract

The reversed shear experiments were carried out on T-10 at the HF power up to 1MW. The reversed shear in the core was produced by on-axis ECCD in direction opposite to the plasma current. There are no obvious signs of Internal Transport Barriers formation under condition when high-k turbulence determines the electron transport.

The reversed shear (RS) experiments were carried out on T-10 at the HF power (P_{HF}) up to 1MW in plasma. The main goal of these experiments was to investigate the possibility of the Internal Transport Barrier (ITB) formation under condition, when the HF power was absorbed by electrons and electron turbulence determines the plasma behaviour.

Nonmonotonous q profiles with RS in the plasma core were created by on-axis ECCD in opposite direction to the initial ohmic current (Counter-CD).

Preliminary calculations were made by ASTRA transport code [1] and by TORAY code [2] for EC-current $j_{cd}(r)$. These calculations were made for the wide range of plasma parameters: total plasma current I_P=75÷150 kA, line average density $\overline{n}_e=(0.7\div1.5)\times10^{19}$m^{-3}, absorbed power P_{ab} up to 1 MW.

It was shown that:

1) It is possible to create the nonmonotonous q profiles with reversed shear area in the plasma core ($r/a \leq 0.35$);
2) the $q_{\min}$ value is changed in a wide range from q_{min}~1.3 up to q_{min}~3 by I_P variation from I_P=150 kA to I_P =75 kA correspondingly.

The T-10 experiments were carried out in the above mentioned range of plasma and HF-power parameters. The toroidal magnetic field B_T provided the exact on-axis power absorption (B_T=2.42Ò). For all cases experiments were made in both Counter- and Co- CD regimes. It allowed to compare a confinement and transport in plasma core under the same conditions but at different $q(r)$ profiles: monotonous (at Co-ÑD) and reversed shear (at Counter-CD).

The following experimental results were obtained:

1. The reversed shear $q(r)$ profile was formed in the experiments.
2. No obvious signs of ITB formation were observed at reversed shear $q(r)$ profile and $q_{\min}^{calc}\approx1.3$ (I_P=150 kA). Central plasma temperatures $T_e^{Counter}(0)$ and $T_e^{Co}(0)$ were close during the whole HF pulse (Fig. 1).
3. When the $q_{\min}^{calc}$ increases to ~2 (at I_P=100 kA, according to preliminary calculations) at the same HF-power the degradation of the central electron temperature $T_e(0)$ (~ 30 %) was observed at about 100 ms after EC-power turn-on (Fig. 2,à). This process was accompanied by development of internal ($r \leq 10$ cm) MHD mode with m/n=2/1 (Fig. 2,b).

4. At the $q_{\min}^{calc}$>2 (I_P=75 kA) the plasma behavior after HF-power turn-on is close to the previous case ($T_e(0)$ degradation with m/n=2/1 mode development), but a new phenomenon was observed. The $T_e(0)$ restoration up to initial (before MHD-phase) level took place after the typical MHD-phase (Fig. 3).

The following peculiarities are essential:

- The $T_e(0)$ restoration is always observed after MHD phase in all examined experiments.
- $T_e(0)$ value on restoration phase does not exceed initial (before MHD) value and remains about 10% lower than $T_e(0)$ at CO-CD (i.e. at monotonous $q(r)$ profile) as it is shown on Fig. 3.
- As it is seen from comparison of $I_{SXR}(r)$ profiles in both Counter-CD (after restoration) and CO-CD there are no obvious signs of temperature gradient increase in the vicinity of q_{min} (Fig. 4).

Therefore, $T_e(0)$ restoration observed in these experiments is not the result of ITB formation.

For more detailed analysis the results of these experiments were examined in simulations by the codes mentioned above. The simulations were carried out in the conditions close to the experimental ones: $n_e(r,t)$ and $T_e(0,t)$ from experiment were used. Dynamics of discharge was taken into account.

The results of the $q(r)$ calculations are shown on Fig. 5,b. The experimental $T_e(0,t)$ trace is shown on Fig. 5,a.

Comparison of the $q(r)$ profiles with plasma behavior in different time moments shows that the $T_e(0)$ degradation in experiment occurs at the moment of the second rational surface q=2 appearance. The restoration phase beginning is in a good agreement with the moment when the q_{min} becomes q_{min} >2 (Fig. 5,a).

This agreement between calculation results and experimental data provides the basis for more probable hypothesis of the restoration phenomenon. When the second rational surface q=2 appears in plasma, an interaction between two islands m=2/n=1 formed near these surfaces can occur. It leads to additional transport losses from plasma core and, therefore, to $T_e(0)$ degradation (which is observed in experiment). The restoration phase begins when q_{min} becomes q_{min} >2 and hence the additional electron transport connected with this MHD activity vanishes.

Analysis of the experimental results has shown that in the q_{min} region, the $E\cdot B$ shear damping rate, $\omega_{E\cdot B}$ is about 3 times lower than the growth rate of low-k ITG modes ($\omega_{E\cdot B} \cong 0.2\times10^5 s^{-1}$ vs $\gamma_{ITG}=0.6\div0.7\times10^5 s^{-1}$) due to low values of the ion temperature ($T_i(0)$@0.3÷0.4 keV), density ($n(0)$@ $3\times10^{19} m^{-3}$) and rotation velocity ($V_\perp \cong 3\div5\times10^3$ m/s).

Moreover, the difference $\omega_{E\cdot B} < \gamma$ is, probably, essentially stronger. Under T-10 conditions ITG mode does not play the significant role because the ion transport is close to neoclassical level [3]. Electron transport is determined by high-k electron modes (ETG) which has larger growth rate [4]. Probably due to this reason the obvious signs of ITB formation were not observed in these T-10 experiments.

The experimental results and the corresponding numerical simulations allow us to conclude:

1. The q profiles with reversed shear area in the plasma core in T-10 experiments have been formed. The possibility to control the current profile by ECCD in the broad range was shown experimentally.
2. The obvious signs of ITB formations in the reversed shear discharges under conditions when the high-k turbulence determines electron transport were not observed.
3. The $q(r)$ profiles with two resonance surfaces are dangerous for plasma confinement in the plasma core when q_{min} becomes close to $q_{min}\approx2$.

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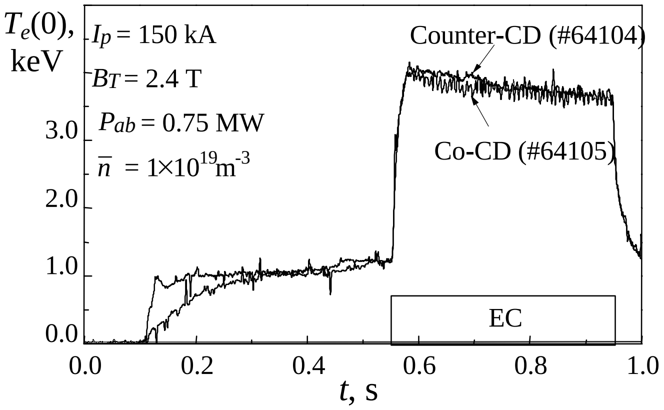

Fig. 1 The $T_e(0)$ traces in CO- and COUNTER-CD discharges at I_p =150 kA

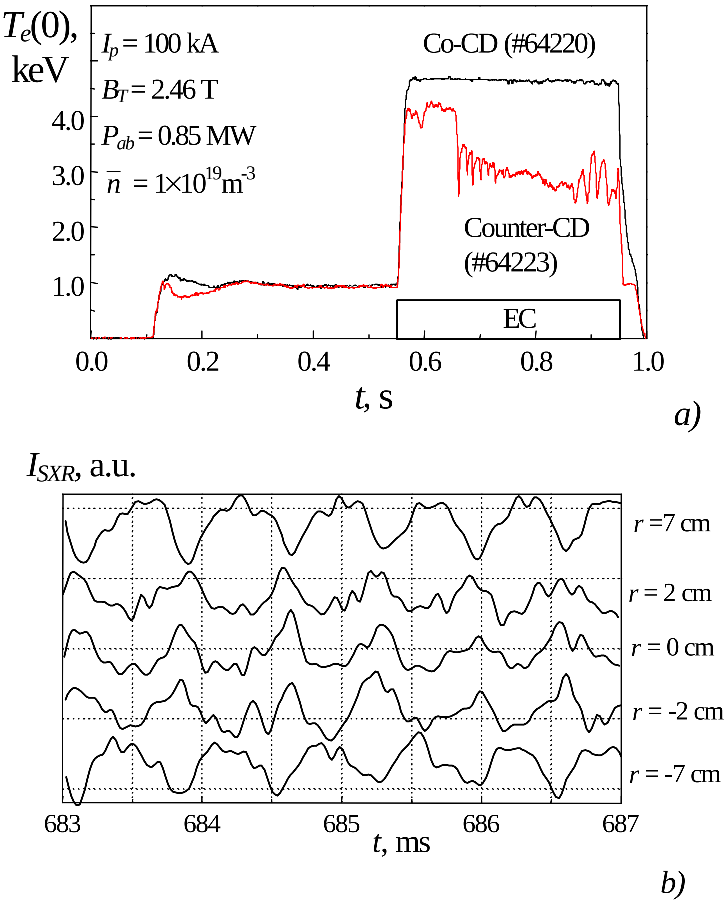

Fig. 2 à) The $T_e(0)$ traces in Co- and Counter-CD at I_p=100 kA; b) The chord soft X-ray intensity I_{SXR} traces.

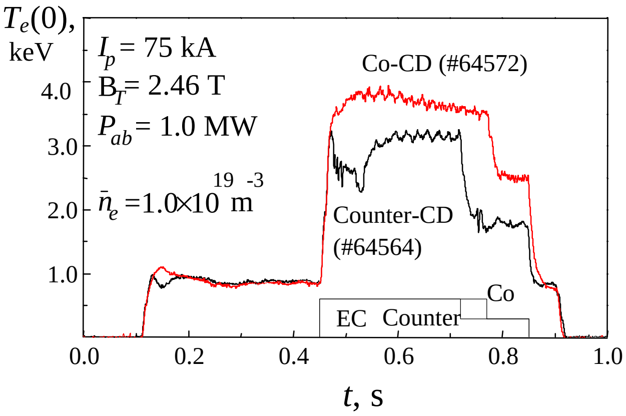

Fig. 3 $T_e(0)$ traces at Co- and Counter-CD at I_p=75 kA

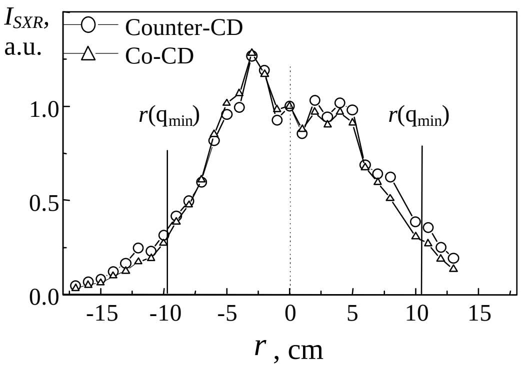

Fig. 4 Chord dependence $I_{SXR}(r)$ for Counter-CD after restoration and for Co-CD

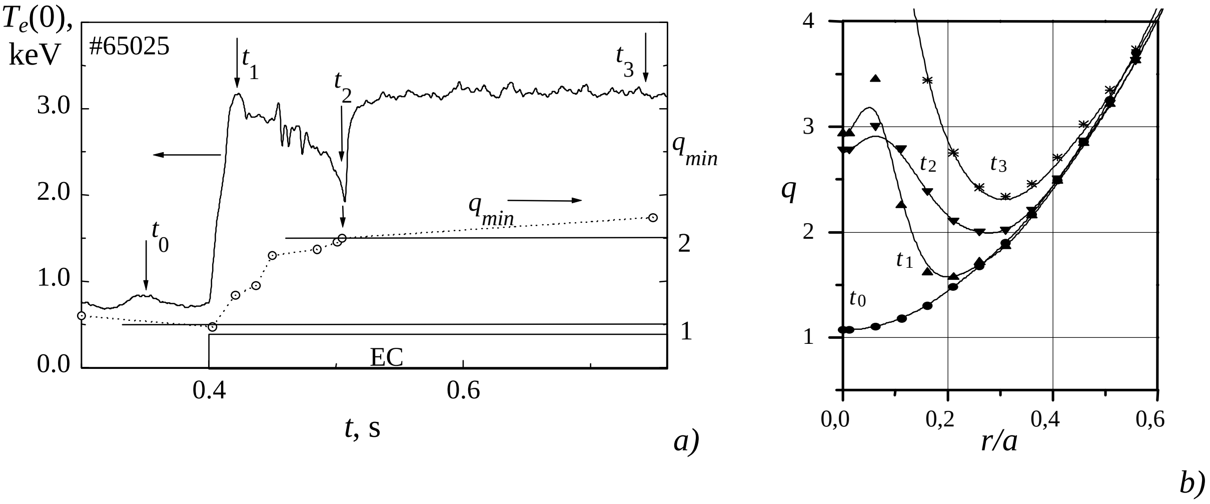

Fig. 5 Experimental Te(0) trace and calculated q_{min} - a)
and results of the q(r) calculations with experimental dynamics taken into account - b).