*35th EPS Conference on Plasma Phys. Hersonissos, 9 - 13 June 2008 ECA Vol.**32D**, P-2.077 (2008)*

Possibility of Scenarios with the Ignition for ITER

V. M. Leonov, V.E. Zhogolev.

NFI, RRC "Kurchatov Institute", Moscow, Russia, e-mail: leo@nfi.kiae.ru.

Introduction. One of the important tasks for the experimental tokamak-reactor is study of the operational scenarios with high Q → ∞ (where Q is the ratio of the thermonuclear power to the auxiliary heating power [1]). Discharges with the auxiliary power equal to zero and sufficiently high fusion power demonstrating the plasma self-heating and energetic efficiency of the reactor (small recirculating power) are usually called as discharges with the "ignition". It would be important to demonstrate in ITER the ignition scenarios with fusion power of 0.5-1 GW with the time duration of several hundred sec > some tens of energy confinement times. For these regimes it is important to study possibility to control the plasma parameters and burning conditions without using the auxiliary power.

As it has been shown by the ITER-team, it is possible to achieve high Q operational scenarios in this device "…if favorable confinement conditions can be achieved" [2]. However, in the experiments it is possible to achieve regimes with the enhanced confinement (compared to the H-mode) mostly in the reversed (or small) magnetic shear (RS) configurations using off-axis auxiliary heating & CD (bootstrap current can not provide the total plasma current). On the first view it looks impossible to achieve the ignition conditions.

In this report it is shown that due to the very long skin time for ITER (time of the current radial profile reorganization > 1000 sec) one can achieve for investigation scenarios without the auxiliary heating (at Q = ∞) with duration of several hundreds of sec and fusion power of order 600-800 MW. For this it is necessary, firstly, to obtain a regime with the enhanced confinement on the initial discharge stage using the auxiliary heating & CD and, secondly, after switching off the auxiliary heating power, to keep this configuration sufficiently long time owing to the skin effect.

To realize current flat-top and burning discharge stage (with Q = ∞) for several hundreds sec it is necessary to form plasma current ramp-up stage with required plasma parameters and radial profiles during the time of the order of several tens sec. Simulations show that using layer accretion method (when plasma current increases simultaneously with increase of plasma size and plasma density) necessary radial profiles can be formed in time ~ 50 sec. In these scenarios requirements to the auxiliary H&CD systems will be reduced (shorter pulse length ~ 50 sec for profile formation and start of ignition). It means that these scenarios can be realized on earlier stage of experiments than long pulse discharges.

Scenario with the ignition. Results of the modelling of the total time dependent scenario with the ignition are presented in **Fig. 1**. The weak negative shear (WNS) configuration with the internal transport barrier (ITB) ($HH_{y,2}$> 1) was considered as the basic one. In these analyses NBI H&CD module of the ASTRA code was used. For LHCD we used semi-empirical approach prescribing absorption width and position (at the outer region, r/a ~ 0.6-0.7) with the CD efficiency of $\gamma_{20} = 0.3$ A/Wm^2. Anomalous transport coefficients were assumed to have parabolic radial dependence $(\chi, D)_{an}(\rho) = k_{an} \cdot (1+3\rho^2)$. Value of the numerical coefficient k_{an} was fitted to obtain necessary enhancement factor for energy confinement time relative to the ITER H-mode scaling. It was assumed that $\chi_e = 0.5 \cdot \chi_i = D_e = D_{He} = D_z$. In the ITB case we supposed that the ion, electron and particle diffusivities were reduced to the ion neoclassical values [3] in the zone of small or negative magnetic shear

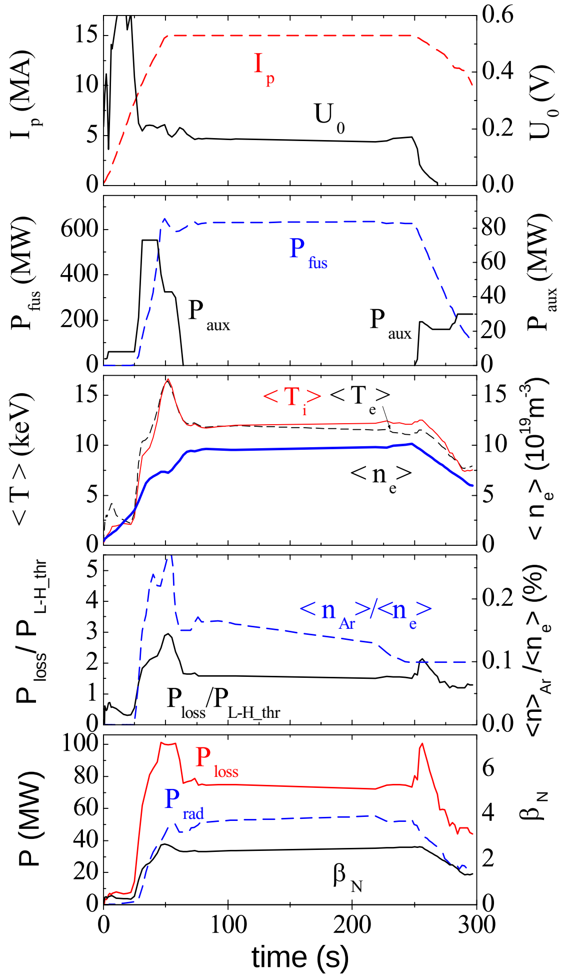

Fig.1 *Evolution of plasma parameters for ITER scenario with the ignition at P_{fus} and P_{loss} control , I_p= 15MA , P_{fus}= 630MW.*

(rq′/q<1) and at the edge transport barrier. The thermalized helium pumping speed was fitted to provide τ_{He} / τ_E = 5 (conservative assumption). Argon seeding fraction which helps to reradiate some power was fitted to keep P_{loss} < 100 MW during the whole discharge (power to the divertor < 30 MW). In this regime it is necessary to have HH_{y2}=1.3. After switching off the auxiliary power the regime with formal Q= ∞ was kept from t~ 60s to the end of the flat-top phase (t~ 250s).

Fig.2 demonstrates profiles for the start (t= 70s) and the end (t= 250s) of the ignition stage. At the start of this stage we have WNS configuration with q_{min} > 2. The value of q_{min} decreases during the discharge and when it approaches q_{min} = 1.1 and sawteeth activity can start we terminate the discharge. Increase in off-axis RF power in the end of the discharge prevents further reduction in q_{min} and rise of l_i and also assists to keep P_{loss}> LH-threshold power during the plasma cooling in the end of the discharge.

As one can see from the **Fig.1** normalized LH-mode threshold power stays > 1.5 during the whole time of the discharge. For this we increased P_{fus} from 400 MW (as in the reference ITER scenario) to ~ 600MW to substitute the 40 MW of auxiliary power by the 200 MW of additional fusion power (40 MW to plasma). This regime is the analog of the reference ITER scenario but with the enhanced energy confinement (HH=1.3) and without auxiliary power.

Decrease in the plasma density and current in the end of the discharge results in fusion power decrease and when this power approaches of about 100 MW (15% from the initial one) we switch-off auxiliary power, make transition to the L-mode, cool the plasma and terminate the discharge.

It is suggested that plasma current and equilibrium conditions are controlled with the feedback systems. Plasma fuelling by gas puffing was used for the control of the average plasma density and thus for the control of fusion power. Simulations show that control of the fusion power by Helium pumping rate is not so effective. To control the

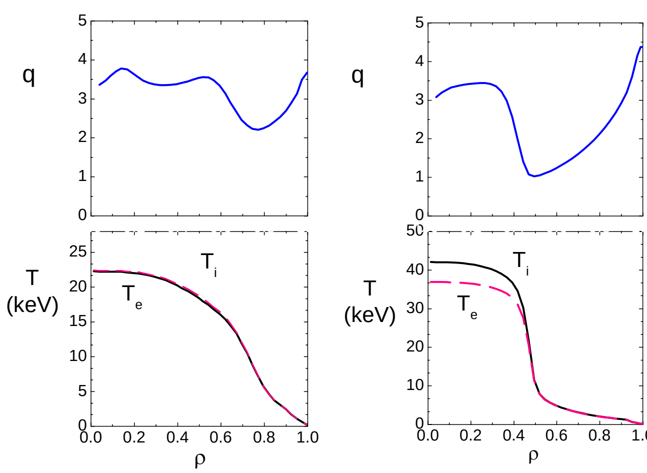

Fig.2 *Radial profiles of plasma parameters at the start (left) and at the end of ignition stage*

divertor heat loads we use Argon impurity seeding (feedback control of P_{rad}) at the level providing $P_{loss} \leq 100$MW during the whole discharge (on the flat-top $P_{loss} \approx 75$ MW). As the main control system we use astatic feedback system [4]. To avoid significant reduction of the fusion power in the time interval when the width of RS zone decreases the feedback control systems provides the rise of plasma density and reduction of the Argon concentration. In the end of the discharge the addition of P_{aux} can increase the divertor heat loads and controlling the Argon concentration keeps power flux to the divertor at the tolerable level below than 100 MW. Value of β_N in this regime is ~ 2.3-2.5, l_i ~ 0.5-0.7 ($4 \cdot l_i$ ~ 2-2.8), the ratio of the average plasma density to the Greenwald density limit is ~ 0.84.

Test of the feedback control systems. To test possibilities of these feedback systems for the simultaneous dynamic control of the burning power and power flux to the divertor we used Carbon impurity injection as a probe perturbation. This perturbation has been added on the steady-state phase of the discharge.

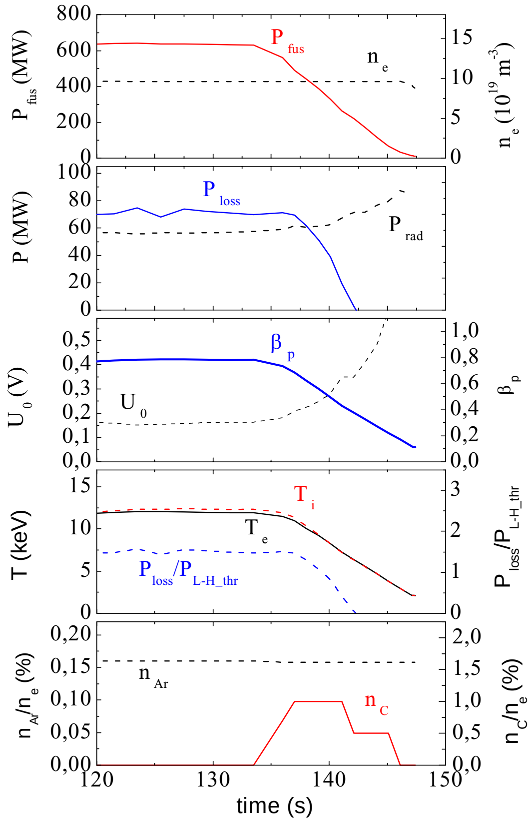

Fig. 3 *Cooling of plasma after Carbon injection without feedback control*

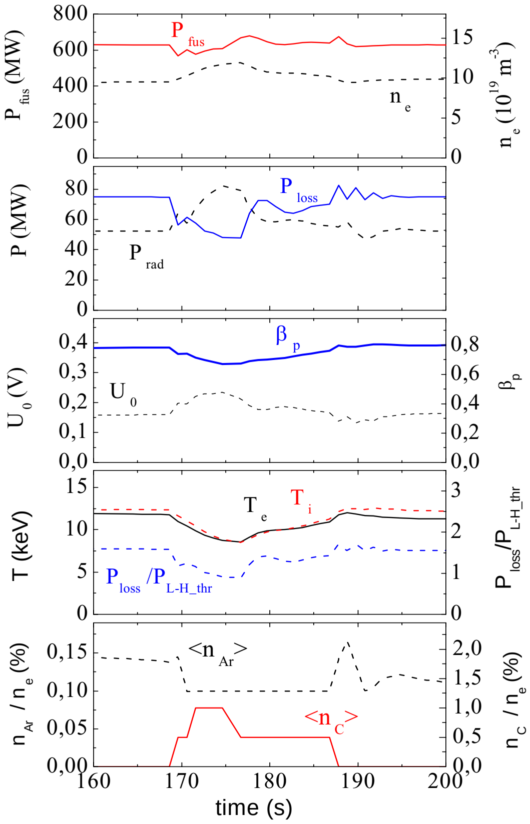

Fig. 4 *Evolution of plasma parameters after Carbon injection with P_{fus} and P_{loss} feedback control*

Fig. 3 demonstrates the case when perturbation (~ 1% sudden increase in Carbon impurity concentration) has been induced on the plasma boundary in regime with open feedback loops (no feedback control). Then this perturbation has been reduced. This perturbation was found to be strong enough to provide complete cooling of the plasma.

Evolution of the plasma parameters in discharge with the analogous perturbation but with closed feedback loops is shown in **Fig.4**. Feedback system changes plasma density to compensate perturbation of P_{fus} and to prevent cooling of the plasma. Other feedback system decreases Argon concentration to prevent strong rise of radiation power and decrease of P_{loss} below than the L-H mode power threshold. After decrease of Carbon influx discharge comes back to the same conditions as before the perturbation. In this scheme of feedback control the change of Argon concentration helps to restore fusion power level also by decreasing of the fuel dilution.

Conclusions:

- **Operational scenarios for ITER with the ignition (Q=∞ and plasma self-heating) are suggested**. It is the long duration of the current radial profile reorganization (>1000sec) that gives a principal idea in obtaining such scenarios. After formation of the necessary radial profiles of plasma current and other parameters with the RS configuration and enhanced plasma confinement in the current ramp-up stage, the auxiliary power is switched off and the discharge continues during several hundreds sec with high P_{fus} and Q value = ∞.

- **Full time dependent scenario** for the discharge with the reference ITER current 15 MA, P_{fus} = 630 MW, HH = 1.3 and burning stage duration ~ 200 sec **is presented**. Simulations for the scenario with I_p ~17 MA and P_{fus} =800 MW have been performed also. Higher plasma current gives possibility to increase the Greenwald density limit and average plasma density for increase of P_{fus}.

These scenarios are based on the H-mode operation with the ITB (conservative view). It is the reason why P_{fus} in considered scenarios is higher than in the reference ITER scenarios (400-500 MW). We must keep permanently the loss power at the level higher than the L-H mode threshold power and compensate the absence of the auxiliary power by the higher fusion power.

- However, in these scenarios the main enhancement factor for confinement can be defined by the ITB in the RS region and **we can operate with the L-mode boundary**. This type of operation looks more perspective because of the working region of plasma discharges can be expanded. It can give the possibility to work with **smaller fusion power** (no necessity to exceed L-H mode threshold power), **without ELMs** what is very important for the divertor operation and with *smaller plasma current*. Besides that, **L-mode_boundary is more compatible with the radiating boundary layer**.

- The next advantage of scenarios with the ignition is the possibility to **use short auxiliary power pulses** (some tens sec) in the initial and the final discharge stages (or in the initial stage only). These scenarios can be used in the early stage of experiments when systems of auxiliary heating would be not fully prepared for the long pulse operation.

- Simulations demonstrate **possibility of the simultaneous control of fusion power**, **plasma density, power losses to divertor** and some other parameters of burning plasma by feedback loops in regime with the ignition **without using of the auxiliary power**.

It is only conceptual view on this subject. It is necessary to optimize these scenarios taking into account the stability analysis of profiles, calculating in more details its working region and so on. If neutron and power wall loading will be too high the discharge duration in first experiments can be reduced.

To verify these ideas it would be interesting to provide test experiments on existing tokamaks (for example on DT JET experiments). To test the possibility of the ITB+L-mode boundary operation with good HH factor (≥ 1.3), possibility of ignition after switching off the auxiliary power at frozen current profile and so on.

This work is supported by Nuclear Science and Technology Department of Rosatom RF and grant SS-371.2008.2.

Reference

[1] ITER Physics Basis Editors et al, 1999 Nucl. Fusion **39,** No 12, 2253.
[2] Technical Basis for the ITER Final Design Report, 1998 Chapter 4, p.2 (Vienna: IAEA).
[3] Chang C.S., Hinton F.L., 1986 Phys. Fl., **29,** 3314.
[4] Leonov V.M., Mitrishkin Yu.V., Zhogolev V.E., 2005 32-EPS(Tarragona, Spain) P5.078