

Cyclic scenarios for steady-state operation of tokamak reactors

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Abstract. A new concept of steady-state scenario for tokamak reactors is proposed. It is based on cyclic operations, alternating phases of positive and negative loop voltage with no magnetic flux consumption on average. Localised noninductive current drive by Electron Cyclotron waves is used to trigger and sustain an Internal Transport Barrier (ITB), whereas Neutral Beam Current Drive is used to periodically recharge the tokamak transformer. The fact of operating in cycles relaxes the hard constraint of simultaneous fusion performance maximisation and full non-inductive operation, within the MHD stability limits. A real time control strategy for this scenario is considered, making use of peculiar properties of the poloidal current density profile. Integrated modelling simulations are performed to apply this concept to the ITER steady-state regime. A linear MHD analysis of the instabilities that could appear in this type of scenario is performed, showing that MHD stability would be strongly improved with respect to a steady regime with a strong ITB.

1. Introduction

A steady-state scenario with 100% of non-inductive current is desirable for the future fusion commercial reactors. However, such scenarios are notoriously difficult to achieve since they require very long pulses with loop voltage practically zero (i.e., no magnetic flux consumption after the initial phase of the discharge). The simultaneous constraints on fusion performance and loop voltage can only be satisfied for very high bootstrap current fractions (higher than 50 % as shown in figure 1), which, in turn, are more likely to be obtained in the presence of an Internal Transport Barrier (ITB), since otherwise very high pedestal heights would be required [1].

Even for ITER [2], the sustainment of an ITB in steady-state scenarios (with a modest goal for the fusion gain, $Q=5$, where Q is the ratio of the fusion power to the additional heating power) can be very difficult if, as expected from present knowledge, it requires strongly negative magnetic shear combined with a large pressure gradient. Note that in ITER, toroidal rotation, which is a main trigger of ITBs in present day experiments, is expected to be low [3], therefore ITBs are expected more in connection with negative magnetic shear plasmas. In such a configuration, the heating system with the highest predicted current drive efficiency, Neutral Beam Injection (NBI), tends to destroy the ITB on the time scale of a few resistive times, due to its broad driven current profile, with a maximum inside the ITB [4]. This is the well known problem of current alignment, which is difficult to control when the bootstrap current is the main current source. Nevertheless, a scenario with only Radio Frequency (RF) heating and current drive systems has been proposed as a conceptual solution to this problem [4]. This scenario relies on the existence of a threshold for the magnetic shear to steadily sustain advanced scenarios. However, the fact that a strongly negative magnetic shear, $s < -0.8$, is needed to sustain such an advanced scenario, and that even more negative values are necessary to attain zero loop voltage, makes this configuration prone to dangerous MHD activity (such as resistive interchange modes), since large pressure gradients are also present at the same radial location.

With the aim of solving these issues, a new type of operation scenario is proposed and analyzed with the CRONOS integrated modelling suite of codes [5] for parameters typical of the ITER steady-state scenario. The basic idea is to alternate between two different states in a

cyclic way. The first state is optimised for ITB establishment and strengthening (at modestly negative magnetic shear and pressure gradient, thus less limited by MHD), but with a non-inductive current fraction $f_{ni} < 100\%$. The second state has very high non-inductive current obtained with NBI current drive and $f_{ni} > 100\%$, which is used to recharge the transformer.

2. The CRONOS code. Models applied

These studies have been performed by means of the CRONOS suite of codes, which can solve the transport equations for various plasma fluid quantities (current, energy, particles, momentum). The modules used can be found in [6], which is the published detailed version of these proceedings.

Since no first-principle model is able to simulate ITB dynamics in present day tokamaks [7], the simple heat diffusivity model $\chi_i = \chi_e = \chi_{i,neo} + 0.4(1+3\rho^2)F(s)$, is adopted where ρ is the normalized radius coordinate, and $F(s) = 1/(1 + \exp(1-s))$. This transport model, although not derived from first-principles, has been already used successfully for the analysis of ITB's in plasmas in JT-60U and in ITER. It is a kind of minimal model, which ensures that the basic phenomena of the suppression of anomalous transport by negative magnetic shear is taken into account. It can be used to reproduce the loss and sustainment of ITB's in ITER by modification of the magnetic shear. For simplicity, the pedestal main features are fixed at $\rho \approx 0.92$ to $T_{ped} \approx 3.5$ keV, which is a conservative value with respect to the scaling laws available for this parameter in the ITER H mode in which the pedestal height roughly ranges from 3 keV to 5 keV. The electron density profile is prescribed with a ramp in the early phase of the regime, then fixed to a Greenwald limit fraction of $f_G = 0.9$. The global parameters used in this scenarios can be found in table 1.

TABLE I: GLOBAL CHARACTERISTICS OF THE ITER CYCLIC SCENARIO

Parameter	Value
Major radius R (m)	6.4
Minor radius a (m)	1.85
Elongation/Triangularity	1.9/0.5
B_t (T)	5.3
I (MA)	8.0
$n_{e,0}/\langle n_e \rangle$ (10^{19} m^{-3})	7.8/7.1
$\langle n_e \rangle/n_{gw}$	0.9

3. Physics basis

The real time control of advanced plasma scenarios is a critical issue due to the stringent conditions for the creation and sustainment of such scenarios. The non-linear interaction between current and pressure profiles leads to a global dependence of the plasma confinement on parameters such as pressure gradient and s/q . Therefore quantities that discriminate the different regimes and which could be used as indicators for real time control are necessary. It has been found in [8] that there is at least a critical physical quantity that discriminates advanced well sustained regimes and H mode scenarios in a global way. This is the poloidal current density, j_θ , which in the plasma core (i.e., except for the pedestal region) is qualitatively and globally different for the inductive H-mode and the noninductive ITB regime, undergoing a global sign change (not only in the reversed q profile - high pressure

region). The Hybrid regime naturally appears as the transition point between the two, with a globally flat and close to zero j_θ profile (not only in the flat q region). In this framework, the issue about the sustainment of ITB's for ITER steady-state plasmas when Neutral Beam Current Drive (NBCD) is added inside the ITB, analyzed in [4], can be explained. In figure 1, the current and electron temperature profiles for the ITER steady-state case proposed in [4] are shown as a function of the square root of the normalized toroidal flux, ρ . In order to study what are the consequences of adding central current, the Lower Hybrid (LH) current drive ($I_{LH} = 0.6$ MA) is removed at $t=1700$ s and replaced by the same amount of on-axis NBCD. In figure 2 the evolution of the q profile and the poloidal current density profile are shown. The q profile changes from highly reversed to monotonic, whereas the poloidal current profile shrinks first towards the center (as long as the ITB shrinks) and becomes positive when the high confinement is lost. The temperature profile evolution, shown in figure 1, also shows how the ITB is lost and the final current density profile distribution shown in figure 2, is clearly dominated by the NBCD. Therefore, in order to control ITB's when NBCD is used and with the aim of minimizing excursions of temperatures or currents, which could lead to the loss of the advanced regime, real time control of the poloidal current seems to be a natural way. This feature is going to be exploited in the scenarios proposed here.

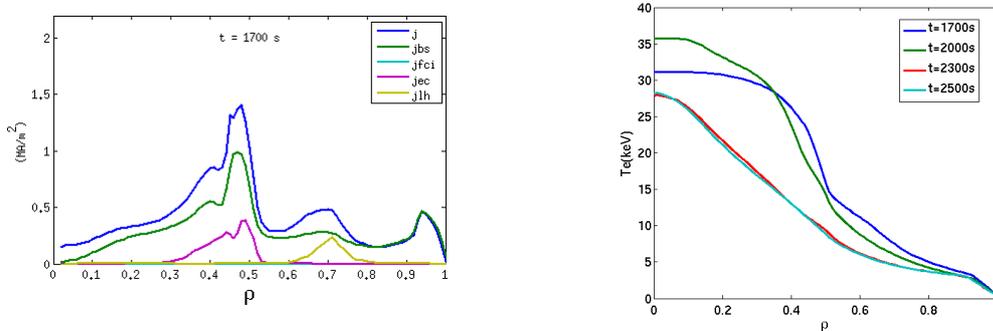


FIG. 1. Total current (j), bootstrap current (j_{bs}), fast wave (j_{fci}), electron cyclotron (j_{ec}) and lower hybrid (j_{lh}) current drive density profiles at $t=1700$ s(a). Time evolution of the electron temperature profile (b).

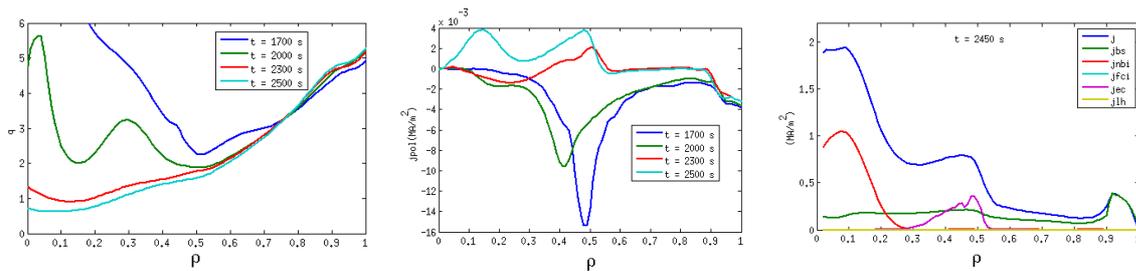


FIG. 2. Evolution of the q profile (a). Evolution of the poloidal current density profile (b). Current density distribution at $t=2450$ s(c).

4. Scenario description

Following the results obtained in [4], the initial phase set-up is configured in order to have a negative magnetic shear close to the threshold needed to sustain such an improved scenario. Since this minimum shear is $s = -0.8$, the shear selected here is -1.0 . For such a purpose, a pure RF scenario without NBCD has been used, with Ion Cyclotron Resonant Heating (ICRH) power $P_{IC} = 18$ MW (53 MHz, 2nd Tritium harmonic), Electron cyclotron Resonant Heating and current Drive (ECRH/ECCD) power $P_{EC} = 17$ MW (170 GHz, O-mode),

$P_{LH} \approx 20$ MW (5 GHz, $n_{||} = 2$). The 17 MW of EC power are deposited at $\rho \approx 0.45$ by using 9 MW from the Upper Steering Mirrors of the Top Launcher at toroidal and poloidal injection angles $\phi_{tor} = 20^\circ$ and $\phi_{pol} = 67^\circ$ and 8 MW from the Upper Row of the Equatorial Launcher at $\phi_{tor} = 38^\circ$ and $\phi_{pol} = 0^\circ$. The total non-inductive current fraction obtained is 81% (with $I_{eccd} = 0.5$ MA and $I_{lh} = 0.8$ MA) and the bootstrap current fraction is 63%. The time evolution of these quantities is shown in Figure 3. The fusion gain obtained in this phase is $Q = 6.3$ with a H_{98} factor of 1.55. With this scheme, the q profile obtained is mildly reversed [2], as shown in Figure 3 with $q_{min} = 2.1$ and $q_0 \approx 3$. Therefore, the temperatures obtained with this q profile do not have strong pressure gradients in the negative magnetic shear region, a favorable circumstance in order to avoid dangerous MHD activity. This global scheme is maintained until the q profile has fully relaxed, which in this case is at $t \sim 800$ s.

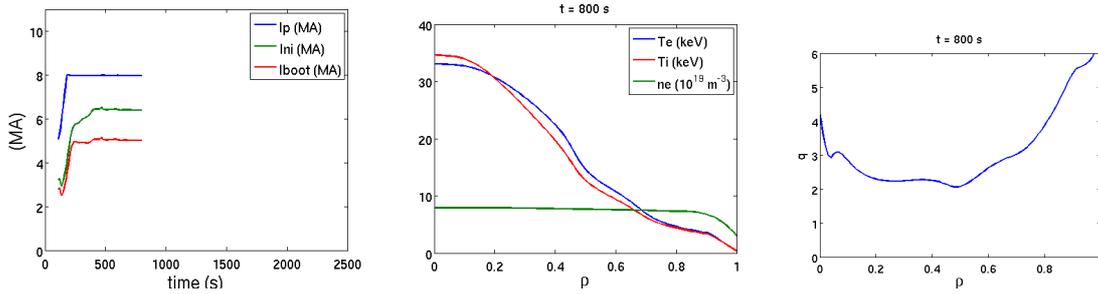


FIG. 3. Evolution of total (I_p), non-inductive (I_{ni}) and bootstrap (I_{bs}) currents (a). Electron, ion temperature and density profiles at $t=800$ s (b). q profile at $t=800$ s (c).

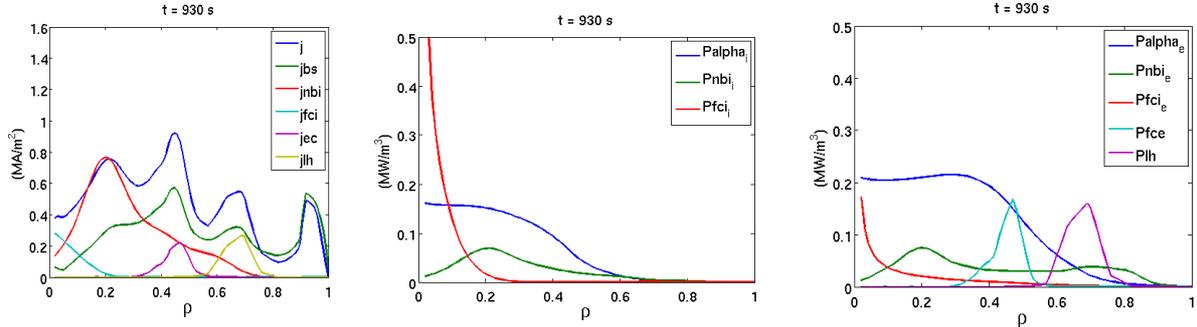


FIG. 4. Total current (j), bootstrap current (j_{bs}), fast wave (j_{fci}), electron cyclotron (j_{ec}) and lower hybrid (j_{lh}) current drive density profiles at $t=930$ s (a). Alpha (P_{alpha_i}), NBI (P_{nbi_i}) and ICRH (P_{fci_i}), power density profiles for the ions at $t=930$ s (b). Alpha (P_{alpha_e}), ICRH (P_{fci_e}), NBI (P_{nbi_e}), ECRH (P_{ec}) and LH (P_{lh}) power density profiles for the electrons at $t=930$ s (c).

From this point, a cyclic regime is established in the plasma by adding 33MW of off-axis NBI heating and current drive (with the geometrical specifications of the ITER injectors [9]) in intervals of 1 min. In addition, 100 kA of an artificial central current (which could be provided by Fast Wave Current Drive with the proper adjustment of the antenna phasing) are also added to control q_0 . In this phase the ICRH power is reduced from 18 MW to 9 MW (in order to prevent excessive growth of the central temperatures) and the power from the ECRH/ECCD upper launchers is reduced from 9 MW to 3.5 MW (in order to avoid an excess of negative magnetic shear at the ITB foot). The aim of this scheme is to keep the main plasma features as stable as possible (fusion power, temperatures, shear) for minimizing thermal excursions on the plasma facing components and current misalignments. As will be shown later, these choices correspond to keep the poloidal current profile as fixed as possible. The current and heating profiles at $t=930$ s are shown in figure 4.

The NBI driven current is 2.6 MA, which together with the increment of bootstrap current obtained (70% in this phase) make the total non inductive fraction to be 120%. The extra current is used to recharge the transformer. In this phase, the fusion gain drops to 6.0, this represents just a 5% drop compared to the previous phase. The reason is that the inclusion of 33MW of NBI power is compensated by the reduction of RF powers and by an increase of fusion power. The plasma performances are in fact better in this transformer recharge phase ($H_{98} = 1.65$), since we now have strong core heating inside the ITB formed in the previous phase. The fact that the poorly current aligned transformer recharge phase can transiently benefit from the high confinement condition created by the ITB optimization phase (because the energy confinement time is much lower than the current diffusion time), is a key ingredient for the average fusion performance of the scenario.

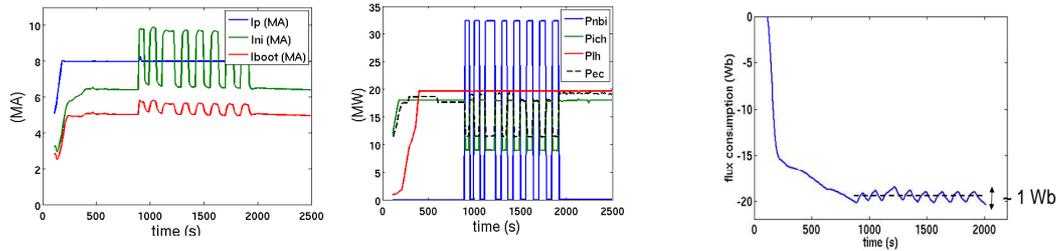


FIG. 5. Evolution of total (I_p), non-inductive (I_{ni}) and bootstrap (I_{bs}) currents (a). Evolution of NBI (P_{nbi}), ICRH (P_{ich}), LH (P_{lh}) and EC (P_{ec}) powers (b). Evolution of the flux consumption (c).

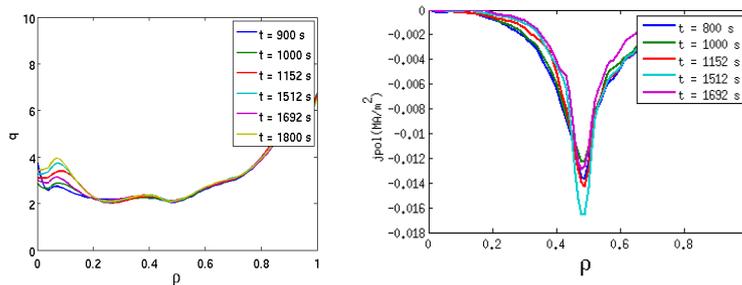


FIG. 6. Time evolution of the q profile during the cyclic phase (a). Time evolution of the poloidal current profile during the cyclic phase (b).

In Figure 5 the time dependence of currents, input powers, fusion power and flux consumption are shown. The total non-inductive current fluctuates between 6.5MA and 9.5MA, whereas the fusion power ranges from 330MW to 420MW. In this cyclic phase, the transformer magnetic flux has also cycles of 1 minute alternating values with differences of 1 Wb around a constant average value. Therefore, the averaged values obtained during this phase are: 0 flux consumption, fusion gain of 6.0, averaged bootstrap current fraction of 67% and averaged $H_{98}=1.6$, which are close to the values obtained for the steady-state scenario discussed in [4], although the H_{98} required in this scenario is lower. However, here the extreme negative magnetic shear, $s=-3.8$, needed to obtain such a high bootstrap current in [4] is no longer necessary. In fact, as shown in figure 6, the q profile remains quite constant with very few variations inside $\rho=0.2$. The variable which has been controlled here in order to prevent plasma excursions has been the poloidal current. As shown in figure 6, unlike in the case shown in figure 2, this current is kept almost fixed in time due to the appropriate choice of power level variations and cycle duration. This could be a basis for the development of an

automatic algorithm for real time control of the scenario using the poloidal current as the main quantity. This would of course require real time equilibrium reconstructions.

Finally, with the aim of analyzing the stability of the scenario without NBI power, the cycles are stopped after 2000s. As shown in Figure 5, the scenario remains stable and close to the initial conditions.

5. MHD stability analysis

The resistive stability issue has been addressed with the CASTOR [10] code assuming an ideal wall at the plasma boundary. The linear growth rate is shown as a function of the toroidal mode number and the MHD displacement associated to the unstable modes are shown in Figure 7. The growth rate γ is normalized to the Alfvén time $\tau_A = R\sqrt{(\mu_0\rho)}/B$, and we plot $\lambda = \gamma\tau_A$. The resistivity η is normalized to $\mu_0 R^2 / \tau_A$.

A series of unstable modes exists for $n \geq 5$. They are localized in regions with negative shear and correspond to Resistive Interchange Modes [11]. These modes have a growth rate that scales with the toroidal mode number as $\lambda \sim n^{2/3}$, and with resistivity as $\lambda \sim \eta^{1/3}$. Their radial structure is very narrow. Below $n=5$, we find no unstable mode.

Resistive MHD stability has been investigated with a fixed boundary, and low- n modes are found unstable as long as the minimum of the safety factor is below 1.98 as shown in figure 7. Above that value, higher- n modes identified as Resistive Interchange modes remain unstable, but their impact on the plasma performance may be less important, due to their much localized structure, as suggested by numerical simulations [12]. It is therefore possible to obtain in the cyclic phase relatively favorable MHD properties, with q_{\min} slightly above 2, as it happens in the reference scenario. This situation is better than for the previous Steady State scenario [4] for which the strong shear reversal allowed in any case the full range of toroidal mode numbers to be unstable. However, since these results highly depend on q_{\min} , a higher value than the one obtained in this scenario would be desirable to avoid being too close to dangerous MHD activity.

This MHD analysis is of course partial, and among the important aspects that are left for future work we shall mention fast particle driven instabilities, which could take their origin in the fusion born alpha particles as well as from heating and current drive systems.

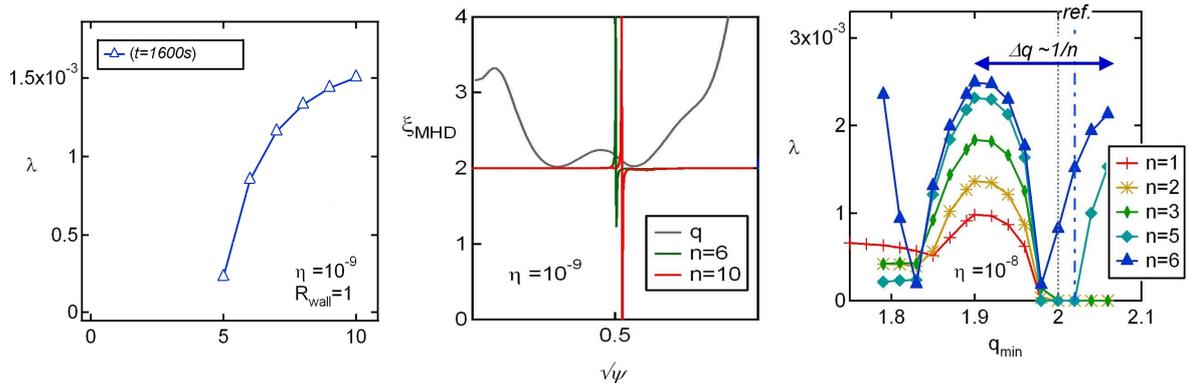


FIG. 7. Linear growth rate as a function of the toroidal mode number. (a) Safety factor profile and MHD displacement (ξ_{MHD}) of unstable modes (b). Role of the q_{\min} value in resistive MHD stability during the cyclic operating regime (the original profile is indicated as “ref.”)(c).

6. Conclusions

A new class of tokamak operation scenarios has been identified, based on the concept of cyclic alternation of different plasma states and the control of the poloidal current density profile. On the basis of a relevant computational example (for the ITER steady-state regime parameters), it has been shown that the fact of combining different states opens up new possibilities to attain plasma performances that would be difficult to get in a single stationary scenario. This technique exploits the naturally long resistive time scales of a burning plasma to decouple constraints that are not easily satisfied simultaneously and to alleviate MHD stability problems. Plasma regimes that are *per se* not suited for steady state turn out to be useful. This fact shows that, the well known problem of having different time scales in plasma physics, which has been regarded as an inherent drawback for the plasma scenarios control, can be, on the contrary, a positive feature if properly exploited. New means for optimisation can then be employed, by varying duty cycle and cycle patterns, as well as by actively using the Ohmic electric field, which would normally be vanishing in a steady-state regime.

Operation of tokamak reactors in burn cycles has been considered in the past (see, e.g., [13-15]). The technique used then was to alternate burn phases with transformer recharge phases without production of fusion power (e.g., at strongly reduced plasma density). The main limitation of that scheme was related to the thermal and mechanical fatigue connected with these drastic changes of the plasma state and energy fluxes, very close to those of pulsed inductive operation. Another possibility that has been suggested and experimentally investigated is the alternating current operation [16-17], with obvious difficulties related to the transition phase in which the current is reversed. The concept proposed here is completely different: the transformer recharge phase is actually the highest performance phase; the main plasma parameters (current, density, magnetic equilibrium) are fixed; the most significant changes are in plasma quantities such as the non-inductive and the bootstrap current fraction; both plasma temperature and fusion power undergo rather modest variations.

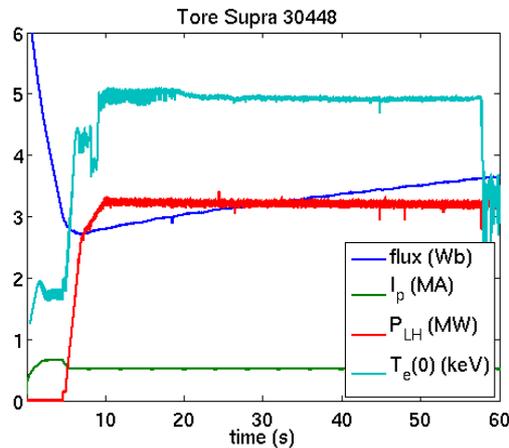


FIG. 8. Time behaviour of edge magnetic flux, plasma current, LH power and central electron temperature during transformer recharge in Tore Supra discharge # 30448.

Transformer recharge on minute-long time scales has been successfully demonstrated in the Tore Supra tokamak, using LHCD to overdrive the current: as shown in Figure 8, magnetic flux variations of the order of 1 Wb have also been induced, in stable plasma conditions. This provides a good basis for an experimental test of this type of regimes. Moreover, the feasibility of cyclic operation on ITER is subject to a number of technological issues, which should be carefully investigated. The most serious one is related to the thermal

cycling of the NBI system, which has been designed so far for < 50,000 cycles, whereas this type of cyclic scenario would typically require ~ 50 cycles for a long pulse of 3000 s. This is not an intrinsic difficulty, but it would probably require some design change. Modulation of the beam energy, without switching off the power, could also be considered. The cycling of the other heating systems should also be studied. Voltage variations in the Central Solenoid associated with the 1 Wb flux oscillations ($\sim \pm 20$ V) should not be a problem; however, additional AC losses in the various tokamak coils have to be carefully evaluated, together with power oscillations in the 69 kV distribution, plasma control at the transitions, thermal cycling on the Plasma Facing Components, neutron flux cycles etc.

Finally, the extrapolation of this concept to the much more stringent constraints of a demonstration or commercial fusion reactor has now to be tested by integrated modelling simulations: since both the useful parameter range and the optimisation knobs have been extended, this will require extensive dedicated studies, which could even result in a set of reactor parameters specifically adapted to the cyclic operation concept.

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