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ITER Plasma Scenarios scaled from ASDEX Upgrade and JET experimental data and their impact on ITER operational space

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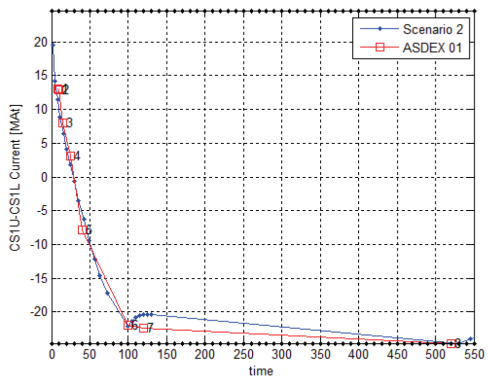
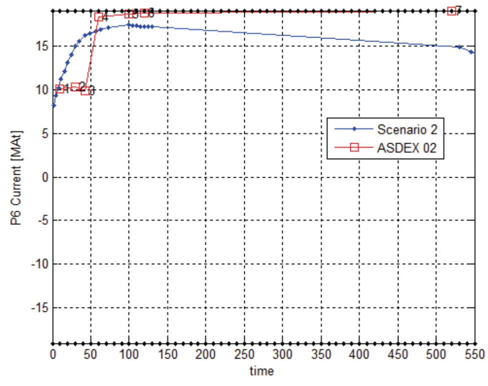
Theoretical predictions and experimental data have been extensively used to establish the range of variation of the ITER plasma scenario parameters [1] since they have a direct impact on the dimensioning of the PF (Poloidal Field) system and on the design of the first wall components. Compared to today's experiments, the working margins adopted for the ITER design are reduced. Resistive flux consumption, *li* and βp ranges play an important role for the capability of the tokamak to guarantee the required plasma shape, plasma current, and flattop duration. Actually the nominal ITER plasma scenario # 2 has an Ohmic slow aperture expansion ramp-up where *li* at flattop is ~ 0.85, and Cejima mean values during ramp up are ~ 0.45 (see ITER design documentation [2], and related documents).

These working conditions are discussed in the light of recent experimental results obtained on JET and ASDEX-Upgrade. Numerical simulations of the ITER current ramp up phase have been produced on the basis of such experimental results for both 15 MA full bore and aperture expansion Ohmic ramp up, which bring to *li* ~ 1.00. On the other hand, an early X-point formation with full bore plasma, offers the possibility to have a heated ramp up allowing lower values of *li* (~ 0.70).

For each type of ramp-up, a suitable number of equilibrium conditions have been obtained optimizing PF currents and plasma shapes with a constrained optimization procedure based on the CREATE numerical codes (see [3] and references therein). The PF system, geometrical and electrical main parameters, limitations on coil currents, maximum fields, vertical forces on Central Solenoid (CS), were taken from the most recent ITER design documentation. The plasma shape was controlled by means of 36 gaps, X-point and strike points positions; the distance between first and second separatrices was kept greater than 40 *mm*, and the plasma at a minimum distance of 100 *mm* from the first wall, according to ITER prescriptions. A bell shaped plasma current profile was assumed without edge current, the internal current distribution parameters being used to fit the prescribed poloidal beta and internal inductance.



ASDEX-Upgrade like ohmic ramp up CS1 current



ASDEX-Upgrade like heated ramp up P6 current

TABLE Ia) Values obtained from 84 Ohmic ASDEX-UPGRADE discharges, ramped to 1MA (at t=1.0) and q_{95}=3.0-3.4
Typically the plasma starts with a limiter phase (~ full bore), is diverted at t=35s.
Maximum and minimum values of the plasma quantities scaled to ITER are provided in parentheses

Ohmic Current rise	**t=15s**	**t=25s**	**t=40s**	**t=100s**
Ip (ITER) [MA]	5.0 (3.3-5.8)	6.75 (6.3-7.0)	9.0 (8.7-9.3)	15 (14.3-15.7)
betapol	0.2 (0.05-0.35)	0.15 (0.1-0.2)	0.15 (0.12-0.18)	0.12 (0.1-0.15)
li(3)	0.65 (0.55-0.75)	0.95 (0.85-1.05)	1.15 (1.1-1.25)	0.92 (0.85-1.0)
Cejima	-	0.4 (0.2-0.5)	0.45 (0.3-0.6)	0.5 (0.35-0.65)
q_{95}	7.0 (6.0-8.0)	5.0 (4.5-5.5)	5.5 (5.0-6.0)	3.2 (3.0-3.4)
a_{min} [m]	0.42 (0.38-0.46)	0.52 (0.50-0.55)	0.55 (0.53-0.57)	0.49 (0.48-0.50)
kappa	~1.15 (1.1-1.2)	~1.2 (1.15-1.25)	1.45 (1.4-1.55)	1.7 (1.65-1.75)

TABLE Ib) Values obtained from discharge 19306, ramped to 1MA (at t=1.0) and q95=3.4.
Typically the plasma starts with a limiter phase (~ full bore), is diverted at t=35s.
Heated with 2.5MW NBI power, L_H transition just after 35 s

Heated Current rise	**t=1.0s**	**t=30s**	**t=42s**	**t=60s**	**t=100s**
Ip (ITER) [MA]	4.0	7.5	9.15	12.0	15.0
betapol	0.14	0.32	0.37	0.53	0.47
li(3)	0.65	1.2	1.2	0.93	0.86
Cejima	0.35	0.50	0.44	0.53	0.51
q_{95}	7.3	5.5	5.1	4.0	3.36
a_{min} [m]	0.485	0.55	0.54	0.49	0.49
kappa	1.20	1.33	1.55	1.77	1.82

TABLE Ic) ITER ramp up values from scaling JET # 70500 snapshot
Current rise to 15 MA in 100 secs, XPF @ 7.5 MA

Ohmic current rise	**t=7.8s**	**t=15.25s**	**t=24.15s**	**t=29.37s**	**t=49.26s**	**t=63.22s**	**t=100s**
Ip (ITER) [MA]	2.5	4.5	6.5	7.5	10.5	12.5	15
betapol	0.056	0.142	0.176	0.143	0.082	0.091	0.079
li(3)	1.03	1.03	1.11	1.17	1.03	0.99	0.99
Cejima	0.45	0.45	0.45	0.45	0.45	0.45	0.45
q_{95}	5.53	4.51	4.63	4.93	3.80	3.34	2.92
a_{min} [m]	1.75	1.83	1.99	1.93	1.97	1.98	2.01
kappa	1.13	1.36	1.49	1.76	1.80	1.81	1.83

Full bore ASDEX-Upgrade like Ohmic Ramp-up

This ramp-up was obtained from the data provided in Table Ia [4]. Most of the reference shapes and geometrical parameter values to achieve a full bore plasma with early X-point transition were taken from the ITER standard Scenario 2 ramp up. The extrapolation of this scenario to ITER reveals the following main issues. The flux consumption in ITER for a *li*~1.00 in ramp up may be critical, since it leaves very little margins for the flat top phase and plasma ramp down. Some of the CS currents, and in particular CS1 (See [2] for coils name definition) reach saturation at the EOB (End-Of-Burn). This implies reduced capabilities of closed loop shape control with the possibility of plasma-inner wall contact near the inner equatorial region. In facts an increase of *li* due to a fast H-L transition or to uncontrolled ELMs may cause a plasma-wall contact. *li* values above 1.0 imply a reduction of the flat top length (~80s for an *li* increase of 0.1) to safely operate. On the other hand the low *li* values observed around 5 MA require high values of P1, P6 and CS3L currents. In particular the saturation of P1 and CS3L may reduce drastically the degree of freedoms for shape variations. Vertical force limits on CS coils are marginally reached. Finally the rapid increase of *li* during the ramp-up (*li*~1.15 at Ip = 9 MA.) may be critical for vertical stabilization.

Full bore ASDEX-Upgrade like Heated Ramp-up

In the heated case studied, an early current rise implies a noticeable increase of *li* up to 1.1-1.2. This increase results from applying heating during the limiter phase (prior to X-point formation) with a significant rise of Zeff. The heated current ramp up was scaled to ITER, see Table Ib. The possibility to heat plasma during ramp up provides lower values of *li* at the start of the flattop. Numerical simulations reveal that for the set of data analyzed, values of 1.1 are critical for vertical stabilization. A strong effort on P1 and P2 currents is required to achieve the desired shape during ramp up. Their saturation means that there are no margins for feedback control.

At flat top, due to heating, *li* is decreased to ~0.7. This may be critical for shape control at the SOF (Start-Of-Flattop) since it requires saturated current and high field values in P6. P1 and P2 current saturation are also observed at SOF-SOB (Start Of Burn).

Due to low *li* values at flat top, the heated scenario assures enough flux at burn also to have an easy feedback shape control at EOB. Due to high values of currents in P6-CS3L, forces in CS3L are close to their maximum allowable value of 75 MN.

Aperture expansion JET like Ohmic Ramp-up

This scenario was scaled from JET pulse #70500 (see Table Ic) which was actually the first dedicated experiment for ITER ramp up designed as an analogue of the old Scenario 2 ramp-up

[2]. The following criticalities emerged in the simulation of such a ramp-up. *li* values turn out to be high during the whole ramp-up. This is critical for vertical stabilization. *li* values around 1.10 at flat top cause more or less the same kind of problems found for the ASDEX-Upgrade like Ohmic ramp up.

Conclusions

A Ohmic ramp up to 15MA in ITER is not realistic within the fundamental machine limits, since the high values of *li* and resistive flux consumption noticeably reduce flat top length. The plasma moves toward the inner part of the first wall and force limits are also reached. Vertical stability problems may arise. On the other hand, if a heated ramp up is implemented, the low *li* values achieved at SOF may bring P6 current in saturation which in turn implies a loss of the desired shape. In particular the distance of the divertor separatrix from the dome may dramatically decrease. Also force limits in CS are reached.

The results obtained so far called for further numerical analyses and experiments [2] aimed at understanding the margins to control *li* and flux consumption values, by exploiting the additional heating systems and the plasma current ramp up rate. A significant amount of work has been carried out by many scientist working on the ITER review design to investigate these problems an also to make sensitivity studies to variations of βp and *li*; sensitivity studies to dIp/dt using transport codes and experiments; studies on the X-point formation; shape optimizations to avoid coils current saturations; studies of vertical stabilization, dome and divertor shape, and of the effect of plasma current profiles shape, especially at the plasma edge, as well as closed loop studies.

The following main design modifications are currently under analysis: increase of the P1-P6 coil maximum currents to enhance closed loop performance and enlarge the ITER operating envelope at low *li*-low flux burned; modifications of the divertor to allow changes in the reference shape; slight variations of the P6 coil position to increase its efficiency at low *li*; use of internal coils for vertical stabilization.

References

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