Controlling ITER Plasma Operation Scenarios

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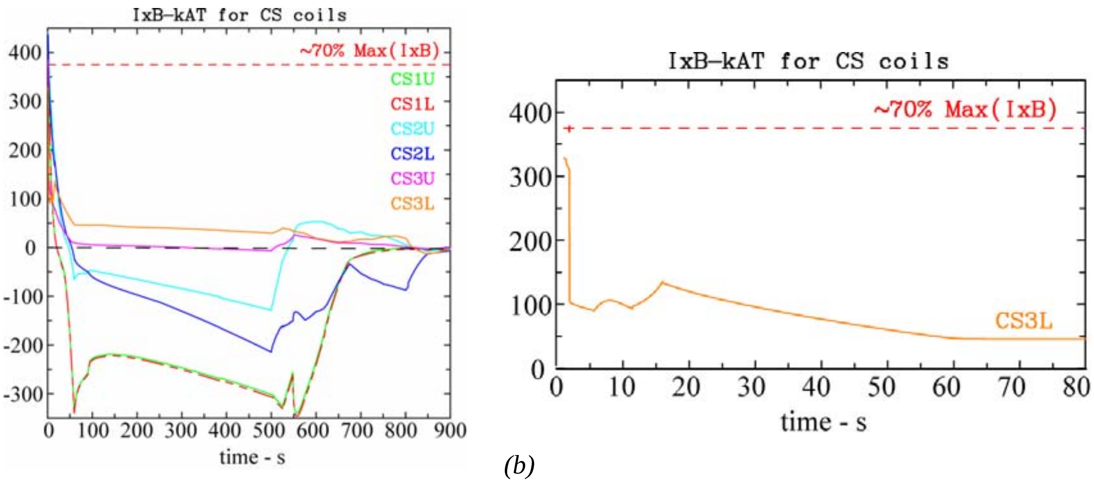
Introduction The ITER Plasma Control System (PCS) is being designed with the aim of conducting the conceptual design review in late 2012. The on-going design effort is being carried out by an international team of plasma control experts from the EU, India, Japan, Korea, Russia, and the US. The design will take into account the requirements for controlling all aspects of the three main operational DT scenarios: the 15 MA inductive scenario for 300 - 500 s duration; the 10 - 13 MA hybrid scenario for up to 1000 s duration; and the 8 - 9 MA non-inductive steady-state scenario for up to 3000 s duration, as well as any special requirements for earlier operation in H, He, and D. Due to space limitations, this paper will describe just a few of the latest control physics issues to arise due to possible constraints on the central solenoid, the in-vessel coils, the electron cyclotron emission (ECE) temperature profile resolution, and the fluctuation in the tritium concentration in the fuel supply.

Magnetic control scenarios with CS3L To avoid schedule delays, the ITER Organization has agreed with the Japanese Domestic Agency (JA-DA) to begin procurement of the bottom module (of 6 modules) of the central solenoid (CS3L) using a superconductor for which the temperature margin was observed to degrade significantly during cyclic testing (over thousands of cycles) to full current and magnetic field conditions. The degradation is believed to be due to fatigue in the superconducting strand and was found to stabilize if the applied Lorentz force on the conductor was reduced by 30%. Some of the most demanding plasma scenarios were therefore analyzed to determine the impact of a 30% reduction in allowed maximum Lorentz force on the superconductor in the CS3L module. Two independent analysis codes were used (DINA, CORSICA) and good agreement was found between them. Results obtained with the DINA code are being presented in a separate paper at this conference [1]. The peak Lorentz force on the CS3L superconductor occurs at breakdown for the highest flattop plasma current and fastest ramp-up scenarios. Fig. 1 shows results of a CORSICA code [2] simulation for a 15 MA inductive > 400 s duration DT burn scenario with a 60 s current ramp-up time starting from 1.2 s. All coils remain well within the allowed limits and CS3L < 70% of the limit. A 17 MA scenario has also been simulated with a 375 s flattop duration that also remained within these coil limits. Within the assumptions made in both the DINA and CORSICA simulations, the primary ITER fusion performance mission goal of maintaining $Q = 10$ for several hundred seconds can be retained even with a 30% reduction in maximum allowed Lorentz force on the CS3L superconductor.



(a) *(b)*

Fig. 1. 15 MA DT scenario simulated with CORSICA using the Coppi-Tang transport model: (a) > 400 s burn duration is achievable even with the 30% reduced CS3L Lorentz force limit; (b) CS3L Lorentz force peaks at breakdown, but remains well within the 30% reduced Lorentz force limit.

Control with internal coils A final decision on the installation of in-vessel coils is expected before the coils' final design review in November 2013. These coils include upper and lower VS coils to increase the operational space for vertical stabilization and 9 sets of upper, middle, and lower coils for ELM control (Fig. 2). The addition of the VS coils would reduce the settling time response for vertical stability control from about 0.5 s using only the external PF coils to 0.1 – 0.3 s with the internal VS coils. The internal VS coils would also increase the maximum vertical displacement of the plasma that could be stabilized from $\Delta Z_{max}/a \sim 0.02$ with the external PF coils to $\Delta Z_{max}/a \sim 0.08$ with the additional in-vessel VS coils, for $l_i(3) < 1.2$ and $\beta_p < 0.65$ [3, 4]. Results from existing machines indicate that $\Delta Z_{max}/a \sim 0.05$ for typical vertical stability control and $\Delta Z_{max}/a \sim 0.1$ for robust control [5]. So, the additional in-vessel VS coils are essential to have robust vertical stability control over a broad range of ITER scenarios, particularly when realistic noise is included.

The in-vessel ELM coils are being designed to provide n=3 or n=4 field perturbations in the plasma edge with values of $|b_r|/B_{T,0}$ up to 6.2×10^{-4} at 90 kAt peak current, predominantly for ELM control. To avoid localized overheating of the divertor tiles, the ELM coil field perturbation will be able to rotate at up to 5 Hz. ELM control with field perturbations has now been demonstrated on DIII-D [6], ASDEX Upgrade [7], and K-STAR [8]. Although some differences are found in these results, it is encouraging that ELM control with edge magnetic perturbations has been observed on multiple machines. Continued R&D includes understanding the role of resonant and non-resonant perturbations in ELM suppression, the effects of these magnetic perturbations on radiative divertor operation, on the mitigated ELM power fluxes, and on particle transport and fueling. Under conditions in which there is headroom in the ELM coil current, the coils may also be used for suppressing resistive wall modes [9], for dynamic error field correction [10], and possibly for plasma rotation control [11]. Thus, the in-vessel coils are an essential set of actuators for many plasma control areas.

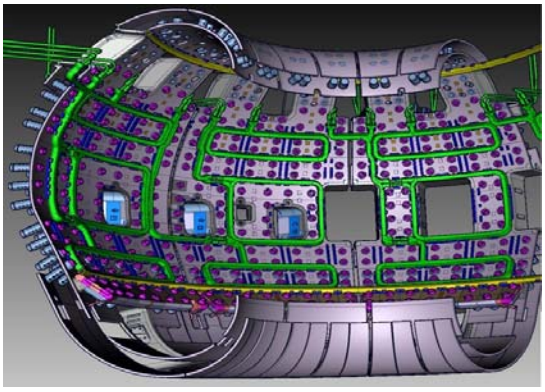


Fig. 2. Cut away 3D view of the inside of the ITER vessel showing the internal coils. The upper and lower VS coils are shown in yellow and three sectors of the 27 ELM coils are shown in green.

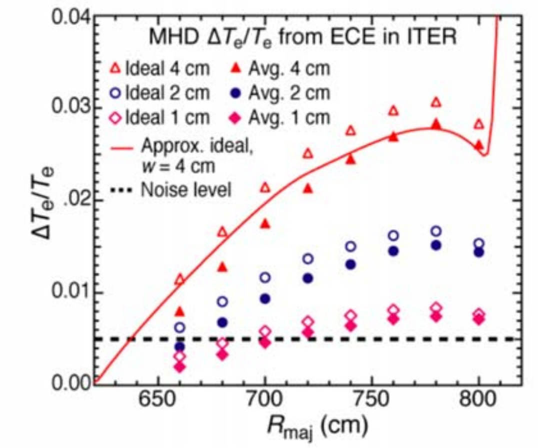


Fig. 3. Calculated relative change in electron temperature due to an NTM vs major radius. Ideal case (open symbols) and relativistic broadened case (solid symbols) for 1, 2, and 4 cm island widths. The dashed curve is the assumed noise limit of the ECE measurements.

NTM control The recent conceptual design review of the ECE diagnostics highlights the measurement requirements for neoclassical tearing mode (NTM) control. The control of NTMs on ITER is essential to achieve and maintain high plasma performance and avoid disruptions. It is critical to be able to measure the NTM island while it is still small enough to avoid mode locking and to lock onto and track the location of the island while using electron cyclotron heating to suppress the mode [12]. However, the high electron temperatures expected in ITER relativistically broaden the ECE, degrading the spatial resolution. Fig. 3 shows the calculated ECE measured relative temperature change due to an NTM in ITER comparing ideal and relativistically broadened measurements for assumed island widths of 1, 2, and 4 cm as a function of major radius [13]. The relativistic broadening degrades the ECE resolution somewhat and the minimum island width measureable is between 1 and 2 cm for major radii where the q=3/2 and q=2 surfaces are expected in the inductive scenario. Since the expected island width when mode locking will occur is ~5 cm, this should be sufficient spatial resolution to measure the island before mode locking occurs. The ECH system requires ~20 ms to begin sweeping the mirror, which can sweep at a rate of ~50 cm/s at mid-radius. This determines how much the mode can grow before locking occurs.

Fusion burn control The detailed design of the tritium plant indicates difficulty achieving a constant 90% T/10% D gas feed to the pellet fuelling system because of the way D and T load into uranium hydride beds during exhaust gas reprocessing and the way they evolve out of them for subsequent discharges. If the T concentration varies widely or gets too low, the fusion burn will be difficult to control and maintain high performance. Initial CORSICA modeling indicates that core $n_T/n_D > 0.8$ is required to maintain high confinement and fusion performance (Fig. 4). At maximum pellet fueling rate (D = 100 Pa m^3/s + DT = 111 Pa m^3/s),

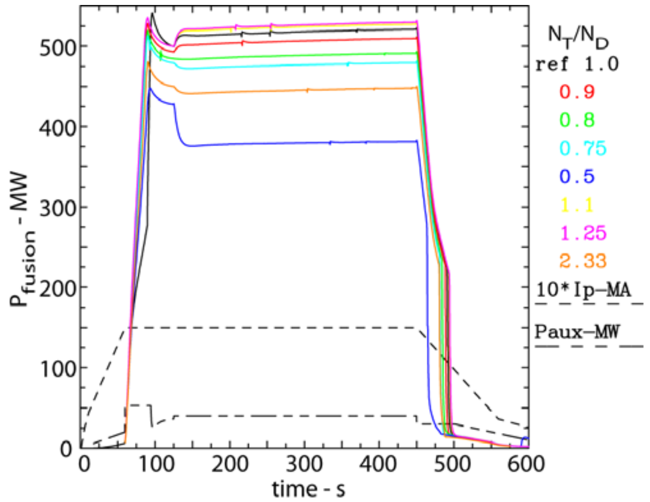


Fig. 4. CORSICA simulations of the fusion power produced for various values of n_T/n_D in the core for the 15 MA inductive scenario with flat density profiles. The drop in fusion power when auxiliary heating is reduced for n_T/n_D < 0.8 is an unacceptable risk to fusion performance.

this implies that the T concentration in the DT fuelling line must exceed 84%, so that the required T concentration in the fuelling line to maintain fusion burn control and high performance is 90% (+10%, -6%). Even for steady-state fuelling within the pumping limit of 200 Pa m^3/s, the required T concentration in the fuelling line must exceed 80%. Although detailed modeling of particle transport, pellet ablation, isotope, and impurity effects is required to verify these estimates, these effects tend to dilute the T concentration in the plasma core and increase the required T concentration in the fuelling line.

Conclusions As the ITER diagnostic and actuator systems begin to be procured, the impact of realistic constraints of those systems on ITER plasma control must be taken into account in the plasma control system design. A few of the constraints on magnetic control, ELM control, NTM control, and fusion burn control have been examined as part of the ongoing PCS conceptual design. A 30% reduction in maximum Lorentz force allowed on CS3L appears acceptable to meet the Q=10 mission goal. The in-vessel VS coils are required for robust vertical stabilization. The ELM coils are valuable for multiple control functions. NTM control appears to be feasible even with the inclusion of relativistic broadening effects on the ECE measurements. The T concentration must exceed 84% for effective fusion burn control to achieve the expected project performance.

References

[1] V. E. Lukash, et al., 39th EPS Conf. on Plasma Phys. (2012) Stockholm, P5.068.
[2] Crotinger, J.A. *et al* LLNL ReportUCRL-ID-126284,1997 NTIS #PB2005-102154.
[3] A. Kavin, K. Lobanov, ITER private communication, ITER_D_33DPE7, April 2010.
[4] G. Ambrosino, et al, ITER private communication, ITER_D_76P4W7, September 2011.
[5] D. A. Humphreys, et al., *Nucl. Fus*. **49** (2009) 115003.
[6] T. E. Evans, et al., *Nucl. Fus*. **48** (2008) 024002.
[7] W. Suttrop, et al., *Plasma Phys. Control. Fus*. **53** (2011) 124014.
[8] Y. M. Jeon, et al., Bull. Am. Phys. Soc., 53rd Meeting of APS Div. of Plasma Phys., (2011).
[9] S. A. Sabbagh, et al., *Phys. Rev. Lett*. **97** (2006) 045004.
[10] Jong-kyu Park, et al., *Phys. Rev. Lett*. **99** (2007) 195003.
[11] A. M. Garofalo, et al., *Phys. Rev. Lett*. **101** (2008) 195005.
[12] R. J. La Haye, A. Isayama, and M. Maraschek, *Nucl. Fus*. **49** (2009) 045005.
[13] M. E. Austin, *Fusion Science and Technology* **59** (2011) 647.