The ITER CS Magnet System

T. A. Antaya¹, J.V. Minervini², J. Feng³, N. Martovetsky⁴, P. Michael⁵, and J.H.Schultz⁶

¹Massachusetts Institute of Technology, Cambridge, MA, <u>antaya@psfc.mit.edu</u>
²Massachusetts Institute of Technology, Cambridge, MA, <u>minervini@psfc.mit.edu</u>
³Massachusetts Institute of Technology, Cambridge, MA, <u>feng@psfc.mit.edu</u>
⁴Lawrence Livermore National Laboratory, Livermore, CA, and <u>martovetsky1@llnl.gov</u>
⁵Massachusetts Institute of Technology, Cambridge, MA, <u>michael@psfc.mit.edu</u>
⁶Massachusetts Institute of Technology, Cambridge, MA, jhs@psfc.mit.edu

ITER is planned as a high Q, extended burn tokamak, with two main experimental goals: (1) a long inductive burn with Q=10 and (2) a steady-state non-inductive burn at Q=5. In addition, the ITER mission includes the integration of fusion technologies, power reactor component testing, and exploration of tritium breeding concepts. As an experimental device with such a broad mission, operational flexibility and extensive diagnostics will be required. To span these burning plasma requirements, active profile control will be required- especially in the gaps g1-g6 in the figure below. Plasma confinement and shaping are largely controlled by adjustment of the toroidal and poloidal field coils and the evolution of the plasma current. The planned long pulse length, of order 300-500s, dictates the use of superconductors in all of the coil systems. To



approach reactor-like conditions, high fusion power (with an associated $\beta^2 B^4$ scaling) is desired. The D-shaped TF magnet coils, shown in elevation, are pushed to the highest possible toroidal field, limited by the maximum allowed conductor field. The TF coils use high performance Nb₃Sn conductors. The outer PF coils, intended to provide plasma shaping and active plasma control, are large and driven more by cost than performance, and will use NbTi conductors. In the inductive heating mode at high Q, the initial heating is obtained by a large flux swing in the central solenoid, and a rapid and complete field inversion from +13.5 T to -12T is required. The CS coils will also employ high performance Nb₃Sn conductors. All coils use cable-in-conduit-conductor (CICC) windings.

On the basis of this key plasma performance role, prior US experience and other factors, the US has proposed to supply all or part of the CS magnet system to ITER, with

the US having the technical lead. The post 1998 ITER rescaling has altered the CS design basis from the tested EDA CS model coil. Recent developments have further changed the design. The CS consists of 6 identical free standing modules, stacked and compressed into a support structure mounted on the upper TF inter-coil cases. The CS no longer bucks the TF in-board legs. Each CS module is assembled from 7 sub-module windings, each of which uses 800m CICC conductor lengths. Module fabrication is wind-and-react followed by impregnation. A number of activities are in progress in the US; centered on baseline design confirmation and risk mitigation, as well as pre-production and industrial scale up planning. The present CS design and performance, key engineering issues, current activities, and the US planning for the supply of the CS are presented.