ITER A Major Step Towards a Fusion Reactor

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Since the beginning of the quest for the peaceful use of fusion energy in the 1950s, physicists have dreamed of producing and studying ignited plasmas, *ie* plasmas in which the power liberated as a result of fusion reactions is sufficient to overcome the power lost by transport and radiation. The path to realizing this dream has been long and arduous, certainly more so than was envisioned by the pioneering fusion researchers in the 1950s. As it turns out, finding a way to the holy grail of ignition has required the development of a new branch of physics, namely the physics of high-temperature plasmas. With the maturation of this field, as evidenced by the impressive successes scored by JET and other large tokamaks of the same generation, fusion physicists at last have the knowledge and capability to realize their dream.

The physical size of a device fully capable of exploring the physics of burning plasmas is large, even somewhat larger than that envisioned for a tokamak reactor producing ~1 GW of electrical power. This means that the cost of the experiment will likely be higher than that projected for a mature tokamak reactor. The problem of the high cost of fusion energy development was recognized in the 1980s and addressed by the formation of an international partnership to design and potentially construct the world's first experimental fusion reactor capable of ignition and sustained burn. The project is called the International Thermonuclear Experimental Reactor (ITER) and the partners are Europe, Japan, Russia and the United States. The primary objectives of ITER are to demonstrate ignition and controlled burn of a DT plasma and to demonstrate technologies essential to a fusion reactor, in particular the operation of breeding blanket modules which would establish the feasibility of tritium self sufficiency.

The ITER collaboration was established in the mid-1980s and has moved through a conceptual design phase (1987-1990) and an Engineering Design Activities (EDA) phase (1992-1998). At the conclusion of the EDA the ITER partners have determined that the design produced during the EDA is technically sound, and if built would likely achieve its mission, but that the cost is too high to proceed to construction. Consequently a three year extension of the EDA is now planned in which a variety of activities aimed at reducing the capital cost and preparing for construction of a reduced cost device will be carried out. Preliminary explorations of reduced cost designs reveal that the essence of the ITER mission can be retained with a device which is scaled down in size to about 75% of that of the EDA design and cost estimates show that such a reduction in size would result in substantial cost savings. As it is premature to elaborate such a reduced cost design, this article will describe the EDA machine and indicate a few of the central physics issues it or its reduced cost successor would explore.

Machine Description

A cross section of the ITER device is shown *opposite* and a list of the main machine parameters is given *below*. From the physics point of view the machine is essentially similar to most presently operating tokamaks, only considerably larger in size (about 2.7 times the scale of JET or 20 times the volume). The toroidal field (TF) is produced by 20 Nb_3Sn super-conducting coils which can generate a maximum field at the plasma centre of 5.7 T, compared with 3.5 T in JET. The poloidal field required to shape the plasma, including the formation of a single null divertor used for heat and particle exhaust, is produced by a set of 9 NbTi poloidal field (PF) coils located outside the TF coils. A Nb_3Sn solenoid which produces 12.7 T on axis when fully energized is used as the transformer to produce and sustain the plasma current. It is situated in the cylinder formed by the noses of the nested TF coils. The maximum toroidal current which can be induced in the plasma is 21 MA, about 3 times the corresponding value in JET.

The use of superconducting coils not only minimizes the power required to energize the machine, it also allows the possibility of reaching full steady-state plasma conditions. The maximum inductively driven pulse length is 1000 seconds, typical of the time required for the plasma current to relax to a steady-state spatial distribution. With the addition of non-inductive methods of driving current in ITER, namely by injection of particle beams and RF waves, the pulse length can be extended and true steady-state burning plasmas typical of those which are foreseen to be in the core of a tokamak power reactor can be produced.

The reference plasma in ITER produces 1500 MW of thermal fusion power, 1200 MW released in the form of 14 MeV neutrons and 300 MW from 3.5 MeV alpha particles. The neutrons escape the confining magnetic field and penetrate the vessel wall where they are thermalized and captured. Most of the alpha power is absorbed by the plasma. About half of this power is ultimately radiated to the first wall while the rest is conducted into the divertor (at the bottom of the plasma chamber in the illustration *opposite*) where it is either radiated or conducted to the divertor plates. The helium 'ash', as the alpha particles are called once they become thermalized, is transported into the divertor and is removed by regenerating cryopumps located in ports near the divertor. Efficient removal of the helium

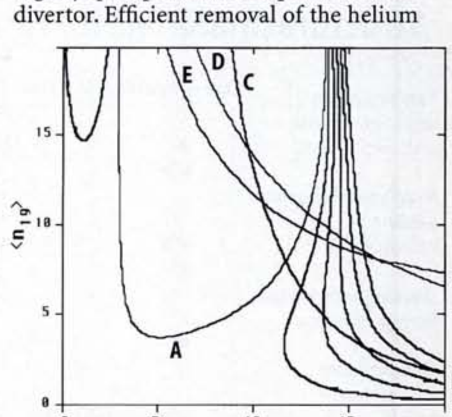

Fig 1 Plasma operational contours in density-temperature space (units of 10^{19} m^{-3} and keV respectively). Curve ***A*** is ignition; ***B*** curves are 20 MW increments of auxiliary power from 20 to 100 MW; curve ***C*** is the L-H transition–operation must be above it; curve ***D*** is the beta limit–operation must be below it; and curve ***E*** is fusion power of 1500 MW

ash is essential to prevent the helium concentration from building up to such a high concentration that the DT fuel becomes too dilute and the fusion 'fire' is extinguished. The operation of the divertor involves a complex interplay of plasma, atomic and surface physics. Heat fluxes of about 5 MW/m^2 on the divertor plates are typical, and values as high as 20 MW/m^2 can be expected under abnormal conditions. The design of a divertor which takes into account the difficult physics and technological constraints was one of the most challenging problems confronted and solved during the EDA.

Operational space and constraints

The operating space of ITER and the constraints which bound it can be conveniently summarized in the diagram shown in *figure* 1. Along curve *A* the plasma is ignited and energetically in equilibrium, *ie* the power supplied to the plasma from alpha particles is equal to the power lost by transport and radiation (including bremsstrahlung and synchrotron emission). The transport losses are modelled in this calculation to be consistent with the global confinement time which is extrapolated for ITER from an extensive database gathered from present day tokamaks. Advantage is taken of the so-called 'H-Mode' or high mode of confinement (*see Campbell, chapter 2.1*) which is predicted to occur only above curve *C*. Along the *B* curves the plasma is also energetically in equilibrium, with the benefit of auxiliary power, however (the curves correspond to 20 MW increments). Also shown are curves indicating 1500 MW of total fusion power (*E*) and the beta limit (*D*), where beta is the value of the total pressure normalized to that of the magnetic field, $B^2/2\mu_0$ (*see Campbell, chapter 2.1*). The operating space for ITER is the triangular region bounded by curves *C* and *D* in the lower right of the figure. An additional limit not shown is a so-called density limit which is simply a horizontal line representing the maximum permissible operating density. Although well characterized, the physics of the density limit is only partially understood. Based on extrapolation from existing devices, it could limit the density in ITER to about 8.5×10^{19} m^{-3}; in this case, ignition would occur only for fusion powers less than about 1000 MW and a small amount of auxiliary power (~50 MW) may be required to reach the design target of 1500 MW.

The operating space indicated in *figure* 1 is not at all unfamiliar as machines such as JET, TFTR and JT60-U have operated at comparable densities and higher temperatures. However, in contrast to these machines the power required to sustain the plasma at these densities and temperatures will be mainly generated by energetic alpha particles produced by fusion reactions. As this power is self- consistently determined by the density and temperature of the plasma the behaviour of the self-heated plasma could be quite different from that of a plasma sustained by auxiliary power sources. Also, the coupling of the energetic alpha particles to Alfvén and other modes in the ITER plasma could drive these modes unstable and lead to a reduction in heating efficiency. The stability of the plasma in the presence of alpha particles, the relaxation of the current profile in the presence of alpha particle heating, the exhaust of the helium ash, and the sustainment of steady-state conditions by non-inductive methods of current drive are all important issues that will be studied in ITER. Successful resolution of these issues is critical to establishing the viability of the tokamak confinement system as a fusion power reactor.

Thus, ITER is the first fusion experiment designed specifically to explore the scientific issues associated with an ignited (or near-ignited) plasma. In addition to enabling fusion scientists to discover and investigate new regimes of laboratory plasma physics, operation of ITER will establish the potential of the tokamak as a power reactor. Unfortunately the time scale for constructing ITER and beginning to extract the physics is rather long, about 10 years. Until then physicists must continue to be content with understanding plasma behaviour in devices with dominant auxiliary heating. But the beginning of ITER's construction will signal a decade-long countdown to the time when the dream of producing and studying ignited laboratory plasmas will finally be realized.

Above ITER as designed in EDA and **below** parameters

Major radius	8.14
Minor radius	2.8 m
Plasma elongation	~1.6,
Plasma triangularity	~ 0.24
Nominal plasma current	21 MA
Toroidal field (at R = 8.14 m)	5.68 T
MHD safety factor	~3
Volume-averaged temperature	12 keV
Volume-averaged density	1×10^{20} m^{-3}
Impurity fractions (Be, He, Ar)	0.02, 0.09, 0.002
Effective charge	1.8
Normalized beta	2.2
Fusion power (nominal)	1.5 GW
Average neutron wall loading	~1 MW/m^2
Burn duration	1000 s
Plasma magnetic energy	1.1 GJ
Plasma thermal energy	1.1 GJ