A COMPACT IGNITION EXPERIMENT*

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Abstract

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A national design team, with representation from essentially all of the major US fusion laboratories, has developed the conceptual design for a compact tokamak device to build on the good results from existing tokamak experiments. This device is called the Compact Ignition Tokamak (CIT), and has the mission of 'realizing, studying, and optimizing ignited plasma operation'. The features of this are compact size, liquid nitrogen precooled copper coils producing about 10 T toroidal field and about 10 MA plasma current at plasma elongations of two, divertor and limiter operation, and ICRF heating.

Relatively small, high field tokamaks as represented by the Alcator devices have been very successful in obtaining excellent plasma parameters with relatively low cost. For some years, it has been advocated that a high density, high field approach would be an effective means for studying the physics of an ignited plasma[1,2]. During the past two years, an extensive study of the physics and engineering issues of this class of devices has been conducted in order to evaluate their feasibility. Based on this effort, we have concluded that a tokamak, the Compact Ignition Tokamak (CIT), with the following parameters, would have the performance necessary to study the physics of burning plasmas:

	Units	Limiter	Divertor
Major Radius	m	1.324	1.339
Minor Plasma Radius	m	0.427	0.412
Plasma Elongation		2.0	2.1
Plasma Current	MA	10	9
Peak Divertor Plate Heat Flux	MW/m^2		5.0
Toroidal Field	T	10.39	10.27
Toroidal Field Flat-Top	s	4.2	
Plasma Burn Time	s	3.6	
Plasma Current Ramp-up Time	s	5.8	
Neutron Wall Loading at 300 MW Fusion Power	MW/m^2	7.5	
Toroidal Field Energy Requirement	GJ	2.8	
Poloidal Field Energy Requirement	GJ	1.7	
Combined Peak Power for TF & PF Coils	MVA	650	
ICRF Initial Installation	MW	10	
ICRF Future Capability	MW	20	
Number of Full Field Pulses		3000	
Number of 70% Field Pulses		50000	

The primary physics requirements for CIT are that it achieve ignition with a burn time of about ten times the energy confinement time for at least three thousand full powered pulses, and be operable with both a limiter and divertor. Achievement of small radius is facilitated by the requirements for pulse length and number of full-power pulses, which have been relaxed relative

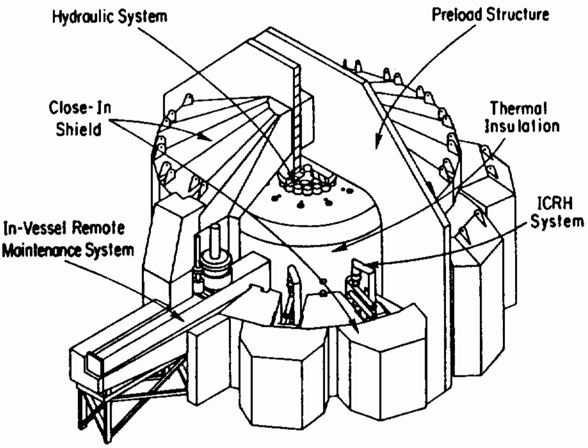

FIG. 1. Isometric of CIT device in its compact cylindrical test cell. Included are the 'igloo' shield and the remote handling mechanism for the hardware internal to the vacuum vessel.

to previously studied fusion engineering test facilities. The relatively short TF flat-top (12 times estimated energy confinement time) can be accommodated by a magnet design in which temperatures rises adiabatically during the pulse. The minimum number of full-power burn pulses has been set to 3.000, but the experiment is capable of operating 50,000 pulses at 70% of the full field and current, limited in a practical sense by the operating span of the machine, which is less than ten years. In order to achieve the minimum major radius, it has been necessary to utilize liquid nitrogen precooling of the magnet in a manner similar to the Alcator-C and the Frascati FT (high-field compact) tokamaks now in operation. The need to recool between pulses, coupled with the short TF flat-top pulse, results in two operational changes relative to the majority of present-day tokamaks; namely, a restriction of one to two full parameter pulses per hour and the requirement to ramp up the plasma current concurrently with the toroidal field. Figure 1 shows the CIT device inside its cylindrical test cell.

The CIT mechanical configuration is based on the following structural approaches, chosen from a number of alternatives studied during the first phase of the conceptual design: (1) use of an external frame and hydraulic press to apply sufficient preload to the TF coil to support the vertical separating force on the TF magnet and thereby minimize the stress in the inner leg of the TF coil; (2) use of wedging in the inner TF legs to support the inward load on the TF coils; (3) use of a gap between the TF coil inner legs and the central ohmic heating (OH) coil, to assure structural independence of the two systems; (4) use of copper-steel laminated conductors in the TF and OH coil in order to optimize the conductivity/strength ratio of the conductors; (5) use of an average stress criterion where the average TF stress is allowed to approach 0.85 times of yield strength of the composite material, in accordance with the guidelines for allowable stress developed for externally supported structures having to perform through a limited number of cycles; (6) use of partial coil cases with significant horizontal access between coils to provide for overall support against out-of-plane loads; and (7) the poloidal field (PF) coil system is external to the TF.

A physics analysis based on existing experimental data has been used to develop the primary physics parameters for the device listed in the table. Particularly important elements of the experimental physics basis used in the design are the Troyon-Gruber[3] beta scaling and the Murakami[4] density limit scaling. A low edge q is specified (2.8) at a level consistent with tokamak experiments to maximize plasma current. The plasma current is further increased by a relatively large elongation, limited to κ=b/a of about 2 by considerations of vertical stability and operational experience on tokamaks. A poloidal divertor and a pellet injection system have been made part of the CIT design to provide a margin for ignition, and to provide impurity and particle control.

The confinement for CIT has been estimated using simple zero-dimensional and sophisticated one and one-half dimensional transport calculations with a wide variety of empirical and theoretical transport scaling formulas[5,6]. One of the major uncertainties of these scalings is the dependence of the energy confinement time on the heating power. To assure maximum conservatism in the machine design requirements, it is assumed that the full alpha heating power is used when the scaling includes a negative dependence on heating power. These estimates indicate that ignition is easily achieved using scalings for the H-mode (Figure 2). H-mode scaling should be applicable since the CIT design contains a poloidal divertor to allow H-mode operation. The ignition margin ranges from acceptable to marginal for the commonly accepted L-mode scalings.

Peaking the profile can increase the ignition margin for a given global energy confinement time, or allow ignition for a lower energy confinement time compared to relatively flat profiles. A significant problem with peaking the profiles is that the q = 1 surface is located at r ~ .6-.7 a, due to the low q at the plasma edge (~ 2.8). Thus, sawtooth oscillations would be expected to flatten the central portion of the density and temperature profiles. Time dependent transport calculations indicate that a combination of off-axis ICRF heating, a carefully tailored current ramp, and pellet fueling can produce a peaked density profile sufficient to allow ignition with even very pessimistic L-mode scalings.

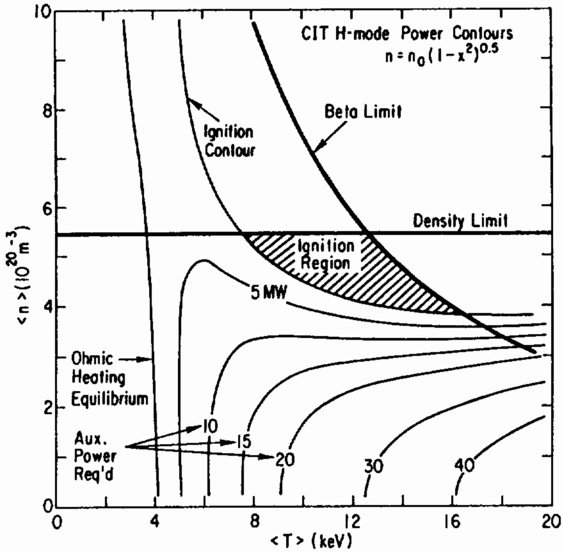

FIG. 2. Contours of constant auxiliary heating power required for steady state operation plotted on a grid of average plasma density and temperature. The two zero power curves are the Ohmic heating equilibrium, and the ignition contour. The operating region is likely to be bounded by a maximum density and beta shown here and described in the text.

The auxiliary heating system is designed to provide the extra heating above the ohmic heating to reach the plasma temperatures required for ignition. The heating systems are to be staged. Initially, 10 MW of ICRF is to be provided. Should that prove insufficuent for ignition, another 10 MW of either ICRF or ECRH will be added. Fast wave ICRF has been chosen for the initial system because it has demonstrated efficient ion heating at high powers and can be implemented in a high density tokamak plasma using rf sources currently available at reasonable cost.

The frequency range has been chosen to be 80-110 MHz. The nominal frequency is 90 MHz for full field operation with minority ^{3}He and second harmonic tritium heating which is the reference heating method. Lower field operation will be possible with minority ^{1}H and second harmonic deuterium heating. The single-pass absorptivity of the two primary modes has been calculated from a one-dimensional mode conversion model[7] to be adequate for efficient plasma heating. The calculations indicate that ^{3}He absorption dominates second harmonic tritium when ^{3}He is present, but even when ^{3}He is absent the residual second harmonic tritium absorption is adequately strong over a broad range of parallel wave numbers.

Wave propagation studies show that the power will be adequately focussed, and Fokker-Planck studies show that there will be relatively little or no high energy "tail" formation due to the high plasma density. The antenna design studies show that the double resonant loop antennas recessed in cavities in the vacuum vessel should be capable of adequate coupling to the plasma at the power densities required for heating the plasma.

Detailed analyses show that operation with an elongation in the range between 2.0 and 2.2 will allow stable operation of a 9 MA diverted plasma. The vertical instability, which has a vacuum vessel stabilized growth rate of about 15 msec, is controlled by two radial field coils located inside the TF. The plasma will evolve from a circular shape at 500 kA to a diverted shape at full current.

Analyses using optimized pressure and current profiles with the PEST code indicate for a triangularity of .5 and elongation of 2, the theoretical beta limit for free boundary, n = 1, external kink modes in high-n, internal ballooning modes is above 10%. This is well above the Troyon scaling value (~ 6%) and provides some margin for practical operation of CIT. Analyses indicate that the peaked pressure profiles produced by strong central alpha particle heating between the sawtooth oscillations are unstable to ballooning modes when a flat current profile is produced by the strong sawtooth oscillations. However, flattening the pressure profile inside the sawtooth mixing radius can produce a stable plasma with high beta's.

The high fusion power of CIT in a compact design poses severe impurity control problems due to the high peak heat fluxes and the large total energy to be handled. The problem is exacerbated by the short radial decay length of .5-.6 cm expected for CIT conditions. The divertor plates and limiters are inclined relative to the flux surfaces to minimize the heat flux, and fabricated using carbon tiles which are not actively cooled. Studies have indicated that the nominal peak heat loads are of the order of 5 MW/m^2.

The double null poloidal divertor was chosen to provide access to "H-mode" confinement and to minimize sputtering of the divertor plates or limiters by operating in the "high-recycling" regime in which a cool, dense plasma is produced by intense, localized recycling and radiative losses near the divertor plates. Detailed calculations indicate that the CIT divertor design can localize the recycling to maximize the probability of obtaining good confinement with the H-mode.

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DISCUSSION

A. GIBSON: What outputs from the present experimental programmes do you regard as essential prerequisites for funding approval of CIT?

J. SCHMIDT: Although we consider the present tokamak experimental programmes as providing important information for developing the physics requirements for CIT, these results are not prerequisites for project funding. A physics research and development plan has been prepared as a guide for the information that would be of most value to the CIT project.