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THE COMPACT IGNITION TOKAMAK PROJECT OVERVIEW*

Prepared for the CIT National Design Team
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Summary

As the next major step in the U.S. fusion program, the Compact Ignition Tokamak (CIT) Project has the objective of reaching ignition in order to address the scientific issues associated with ignited plasma regimes. The present level of uncertainty in plasma confinement scaling requires that the CIT have a high design margin to ensure ignition. The need for adequate margin, coupled with declining budgets in the U.S. fusion program, requires both conservatism and flexibility in the development of the design and operating parameters for the device. To accomplish this, the design includes the provision for an upgrade in performance, which will be partially built into the initial machine installation at Princeton Plasma Physics Laboratory (PPPL).

Design Strategy

In the last decade, several proposals (Coppi,[1] Conn et al.,[2] and, more recently, Schmidt et al.[3]) have advocated small, compact, high-field copper magnet tokamaks as a means of obtaining sufficient confinement for ignition in a relatively low-cost machine design. PPPL has developed a conceptual design of the CIT,[4] which, when coupled with existing facilities at Princeton and other existing equipment in the fusion program, can be built at a fraction of the cost of more conventional larger machines, which rely more on plasma size for ignition.

The difficulty with the compact machines is in the thermal and structural design problems that result from the high-field operation. Magnetic field in the toroidal field (TF) coils of a tokamak results in an outward loading similar to that of a pressure vessel. The equivalent pressure loading vs magnetic field is shown in Fig. 1. The figure also shows, for purposes of comparison with CIT, the magnetic field values for two currently operating large tokamaks, the Tokamak Fusion Test Reactor (TFTR) and the Joint European Torus (JET). Note that the pressure level for the CIT is at least three times that of existing large tokamaks designed to operate with deuterium-tritium (D-T) fuel. Thermal loading on the first wall and divertor plates is likewise strongly influenced by the magnetic field. Figure 2 shows the plasma power density vs magnetic field for the CIT design assumptions. Note that the CIT value is more than three times that of power reactor designs such as the International Tokamak Reactor (INTOR) and Starfire. The CIT design is thus very much driven by thermal and structural engineering design requirements.

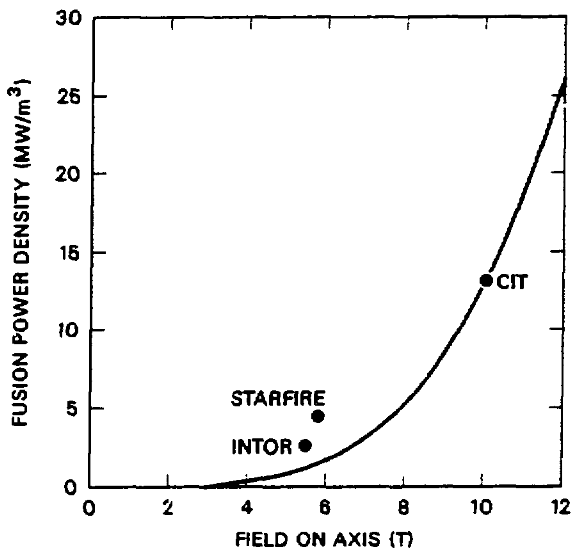

Fig. 2. Fusion power density vs field on axis.

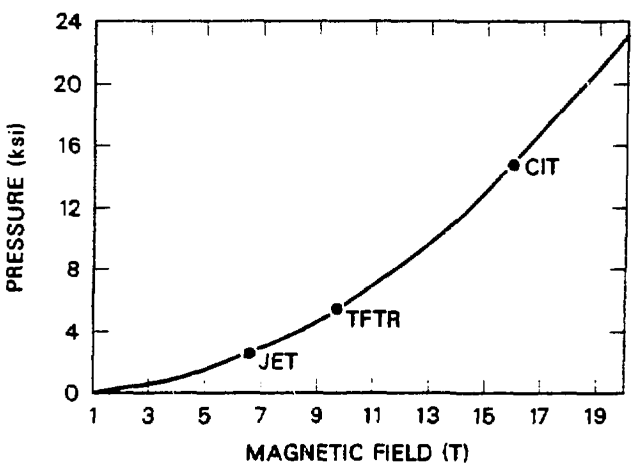

Fig. 1. Magnetic field vs pressure.

The physics/operating requirements have been selected to ensure ignition. In the past year, the major radius described in the CIT Conceptual Design Report[4] has grown by about 0.5 m (from 1.22 to 1.75 m). The bar chart in Fig. 3 shows the individual changes that have occurred, resulting in the total growth. The plasma configuration has been optimized for ignition in a compact size to include all of the goodness factors indicated by recent tokamak confinement experiments.[5] The plasma features high field and high current in a highly elongated cross section with a double null poloidal divertor.

In the event that the CIT fails to achieve the necessary confinement for ignition, even under the rather conservative assumptions used in the present design, the project is providing the capability for a machine upgrade. A preload structure now being designed can be installed at a later date to allow a 20% increase in the magnetic field. Additional machine and facility modifications such as additional power supplies will also be required.

Design Description

Selected machine parameters are shown in Table 1. The initial operating values are consistent with the project funding level; the upgrade values are for increased performance, if required later, by means of increased magnetic field.

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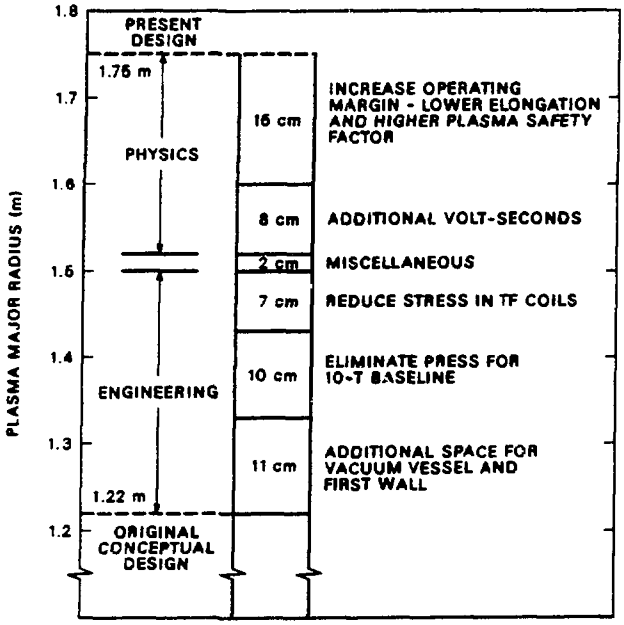

Fig. 3. Increase in major radius since original conceptual design.

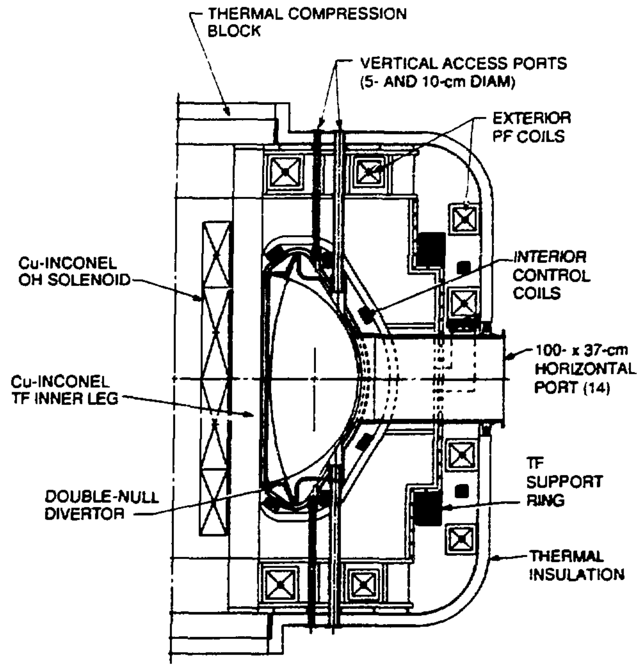

Fig. 4. Elevation view of the CIT device.

Table 1. Selected CIT parameters

Parameter	Initial	Upgrade
Major radius (m)	1.75	1.75
Minor radius (m)	0.55	0.55
Plasma elongation	1.8–2.0	1.8-2.0
Plasma current (MA)	9.0	9.0
Plasma safety factor	3.5	4.1
Field on axis (T)	10.0	12.0
Plasma burn time (s)	7.0	5.0
Plasma heating power (MW)	10	20
Fusion power (MW)	300	
Full power pulses	3000	

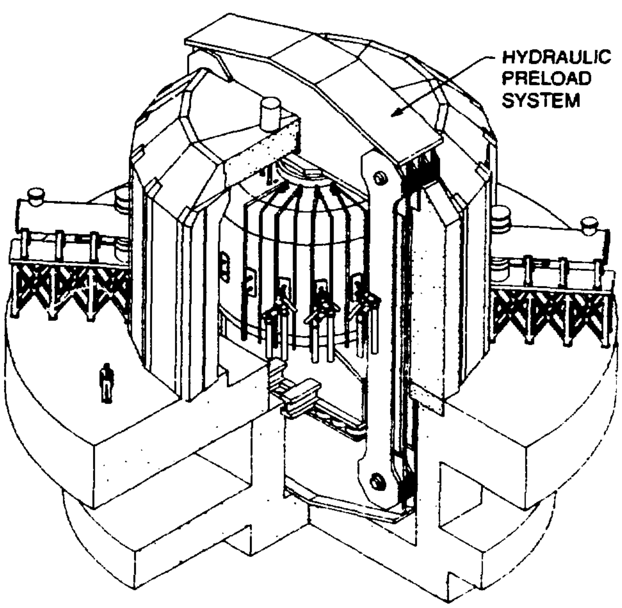

Fig. 5. CIT device with hydraulic preload system.

An elevation view of a cross section of the tokamak device is shown in Fig. 4. The poloidal field (PF) system, mostly external to the TF coils, has been optimized to minimize energy requirements. The separatrix and poloidal divertor shape forms almost naturally. The close-fitting vacuum vessel and TF coils are other distinguishing features of the compact design.

A close-fitting igloo shield is used to limit neutron-induced radiation. The igloo is sized to permit personnel access inside the test cell after several days of shutdown. It also limits air activation in the test cell by providing a sealed compartment for an inert gas in the high-neutron-flux region.

The preload system, shown in Fig. 5, consists of a hydraulic press sized to relieve the tension on the inner legs of the TF coils for the high-field design upgrade. The required preload exceeds 55,000 t.

Magnet System Design

A unique TF and PF coil structural design concept now under development consists of a high-strength, copper-Inconel composite material. A cross section of the TF magnet composite structure is shown in Fig. 6. The high-strength Inconel plate material is in a direction normal to the primary coil loading, which is hoop compression since the coils are wedged. The Inconel material reinforces the thin copper plates through transverse shear. The PF coil plates, shown in Fig. 7, provide reinforcement in the same direction as the primary hoop tension. Montgomery et al.[6] have argued that the composite material, along with the wedged TF design, provides a more efficient design than other available options. An extensive analysis and test program is underway to develop and verify the design.

Vacuum Vessel, First Wall, and Divertor Design

Under ignited conditions, the CIT will produce 300 MW of fusion power for a few seconds. Since 20% of the fusion power is in the form of fast alpha particles, 60 MW of power will be deposited on the first wall and divertor. The radiation fraction

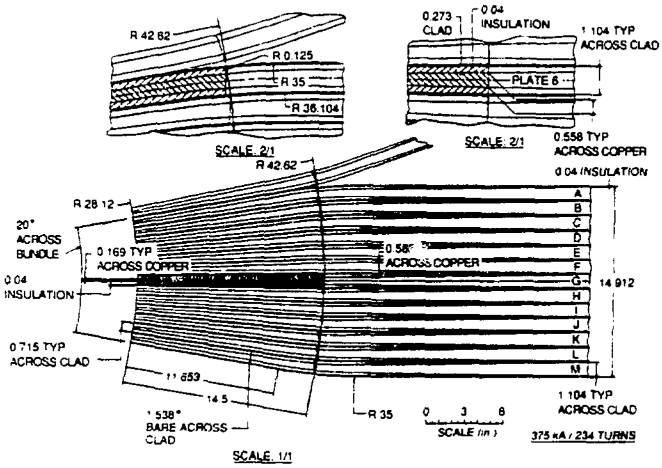

Fig. 6. TF conductor design details.

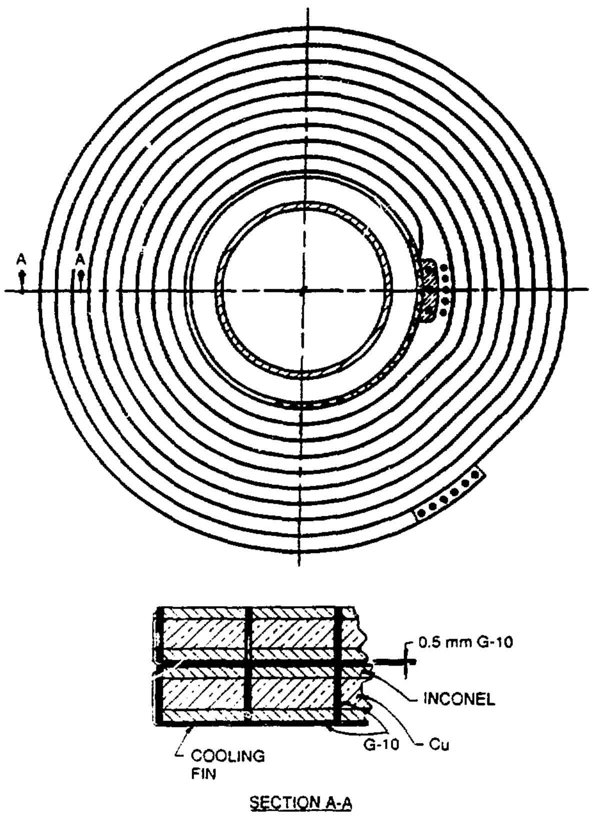

Fig. 7. Typical PF solenoid pancake coil (plan view).

amounts to ~12 MW, leaving 48 MW to be delivered to the two divertor chambers. The compact vacuum vessel configuration limits the available collector area to 1–2 m^2; therefore, the average heat deposition is 25–50 MW/m^2. With no provision for active cooling, graphite tiles appear to be the only viable material that can withstand the 3000 full-power pulses.

Local hot spots caused by nonuniform heat deposition further increase heat loading to the point where local failure would occur in only one burn pulse. A new concept is under investigation to continuously move or "sweep" the location of the divertor strike point during the plasma pulse. Thermal-mechanical and electromechanical concepts have been described by Leuer et al.[7] and Strickler et al.[8] The elevation view of the vacuum vessel and first wall shown in Fig. 8 illustrates the motion of the strike point along the collector plate.

The maximum heat load on the first wall occurs when the plasma is operated in a nondiverted or limiter mode. In this design condition, the 48 MW of power that would otherwise go to the divertor is deposited on the first wall. Because a much larger area is available at the first wall, the maximum heat flux is on the order of 5–10 MW/m^2 and is much easier to handle than in the divertor case.

The first wall and vacuum vessel designs are also strongly driven by off-normal conditions caused by plasma disruptions. Structural loads on the vacuum vessel have been calculated by Sayer using the tokamak simulation code (TSC) developed at PPPL by Jardin. Preliminary results indicate a potential buckling problem caused by inward pressure loading of up to 0.7 MPa at the midplane of the vacuum vessel. Figure 9 shows the pressure distribution for a disruption with an inward-moving plasma for a current ramp-down rate of 2.6 MA/ms. Characterization of plasma disruptions is currently the subject of a major national research effort.

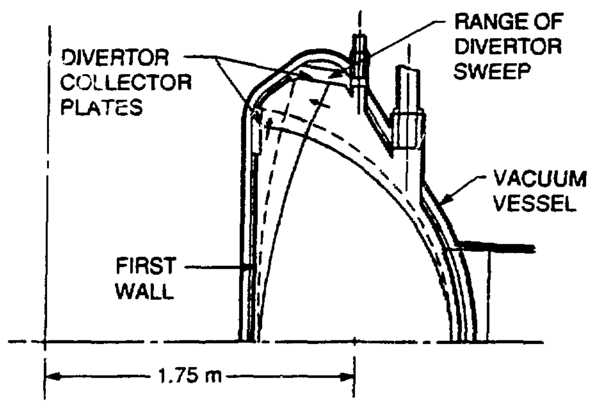

Fig. 8. CIT vacuum vessel first wall and divertor.

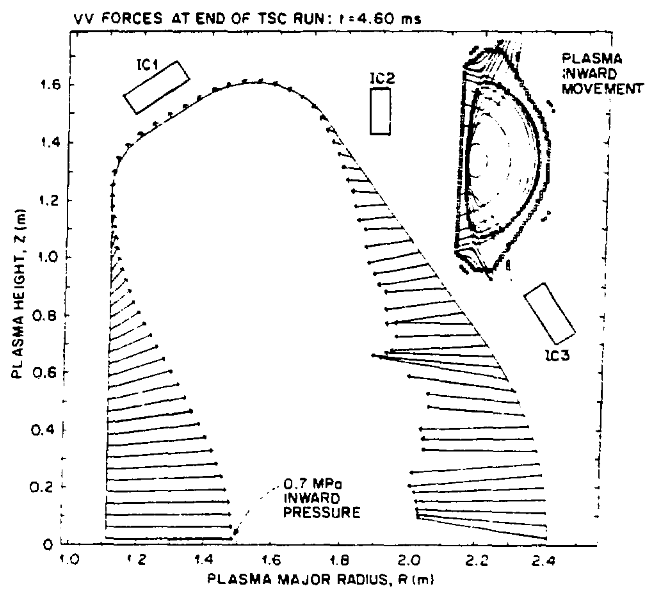

Fig. 9. Pressure loading resulting from inward-moving plasma disruption.

Project Implementation

The CIT is a focused national effort involving the coordinated resources of a large part of the U.S. fusion program. Although PPPL has been designated as the overall project integrator and will be the site for construction of the CIT, project participants include many of the U.S. fusion research laboratories. Additionally, there will be one major industrial participant for the vacuum vessel system.

The project is scheduled as a Department of Energy line item capital project beginning in fiscal year 1988. First plasma operation is scheduled for 1993.

References

[1] B. Coppi, *Comments Plasma Phys. Controlled Fusion*, **3**, 47, 1977; R.L.E. Report PRR-75/18, Massachusetts Institute of Technology, 1975.

[2] D. R. Cohn, R. R. Parker, and D. L. Jassby, *Nucl. Fusion*, **16**, 31, 1976.

[3] J. A. Schmidt and H. P. Furth, *Compact Ignition Tokamak Conceptual Design Report*, A-860606-P-01, Princeton Plasma Physics Laboratory, June 6, 1986.

[4] J. A. Schmidt et al., in *Plasma Physics and Controlled Nuclear Fusion Research 1986 (Proc. 11th Int. Conf. Kyoto 1986)*, IAEA, Vienna, 1987.

[5] R. Parker et al., "Physics Considerations for the Compact Ignition Tokamak," paper presented at the 14th EPS Conference on Controlled Fusion and Plasma Physics (Madrid, June 1987) (to be published).

[6] D. B. Montgomery, J. Chen, and H. Becker, *Structural Support and Conductor Considerations for CIT*, CIT Engineering Report G-870810-PPL-01-ER, May 1987.

[7] J. Leuer, R. Stamburgh, and J. Wesley, "A Magnetically Swept Open Divertor Concept," paper to be presented at this symposium.

[8] D. J. Strickler, S. Jardin, Y-K. M. Peng, and R. Sayer, "Poloidal Magnetics and Divertor Strike Point Control in the Compact Ignition Tokamak," paper to be presented at this symposium.