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OVERVIEW OF THE COMPACT IGNITION TOKAMAK*

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ABSTRACT

A national team has developed a baseline concept for a Compact Ignition Tokamak (CIT). The CIT mission is to achieve ignition and provide experimental capability to study the behavior of burning plasmas. The design uses large magnetic fields on axis (about 10 T) and large plasma currents (about 9-10 MA). The magnet structure derives high strength from the use of a copper-Inconel composite plate design in the nose region of the toroidal field (TF) coil and in the ohmic heating solenoid. Inertial cooling is used; liquid nitrogen temperatures are established at the beginning of each pulse. Capability is provided to operate either with a divertor or limiter based plasma. The design is very compact (1.32-m major radius, 0.43-m plasma radius), has 16 TF coils, and has 16 major horizontal access ports, about 30 cm by 80 cm, located between TF coils. The schedule is for a construction project to be authorized for the period FY 1988-93.

1. INTRODUCTION

Tokamak design studies in the United States have been performed for several years to develop a next-generation tokamak to ignite, to provide a long-pulse burning plasma, and to provide engineering development to form a basis for a demonstration reactor.

During 1985, studies of very compact ignition devices were undertaken; these devices had major radii in the 1.1- to 1.5-m range. A dominant factor encouraging the focus on compactness was the capital cost to design and construct such a fusion device. These studies were performed by a team that focused on Ignitor [1], mostly from Europe, a team at the Massachusetts Institute of Technology (MIT) that focused on the LITE [2] concept, the Ignition Studies Project [3,4] team at Princeton Plasma Physics Laboratory (PPPL) that focused on several concepts, and a team at the Fusion Engineering Design Center (FEDC) [5] that focused on a U.S. version of the Ignitor ideas.

In 1985, a review by an international panel concluded that these compact design approaches appeared feasible and that continued study was warranted.[6] In early 1986, a

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national design team was formed in the United States to narrow the focus and develop a baseline concept for a conceptual design effort. The project team is directed by PPPL with participation of many of the U.S. fusion institutions. A baseline concept was selected in early 1986; conceptual design is in process.

2. DESIGN DESCRIPTION

The configuration of CIT is shown in elevation and plan views in Figs. 1 and 2. Key parameters are given in Table I.

Table I. Key Parameters of Compact Ignition Tokamak

	Divertor	Limiter
Major radius, m	1.34	1.32
Minor radius, m	0.41	0.43
Elongation	2.4	2.0
Field on axis, T	10.3	10.4
Plasma current, MA	9.0	10.0
Fusion power, MW	300.0	
Neutron wall load, MW/m^2, average	6.2	
ICRH power, MW	12.0	
Pulse flat-top time, s	~5.0	
Total volt-seconds, V-s	~30.0	

Distinguishing features of the design, as shown in Figs. 1 and 2, are the following:

- an external frame structure in combination with a hydraulic press system is used to apply a preload to the inner leg of the toroidal field (TF) coil to accommodate the vertical separating force and thereby minimize the induced stress;
- explosively bonded laminated copper-Inconel sandwich plate material is used as TF and ohmic heating (OH) conductors to improve the conductivity/strength ratio of the conductors;
- wedging of the inner legs of the TF coil is used to react the centering force on the TF coils; the TF coil inner legs and the central OH solenoid coil are structurally independent (i.e., no bucking between the systems);
- the major poloidal field (PF) coils are located external to the TF coils; several single-turn control coils are positioned internal to the TF coils;
- an average stress criterion is used; the average TF stress is allowed to approach 0.85 of the yield strength of the laminated material. This is based on guidelines [7] established for allowable stresses developed for externally supported coil structures designed to operate for a limited number of cycles.

The preload structural frame and associated hydraulic system are key configuration features that distinguish CIT from current tokamak devices. The preload structure reacts the precompression load using a hydraulic system that applies a preload of 222 MN (50 Mlb) to the inner leg of the TF coils. A cryostat establishes a thermal boundary between the liquid-nitrogen-cooled tokamak "core" and the "warm" (room temperature) hydraulic system and associated frame structure. The frame structure consists of an array of two stacks of welded Nitronic 33 stainless steel plates; each stack consists of 12 plates (2.5 cm each) bolted together to form one effective plate assembly.

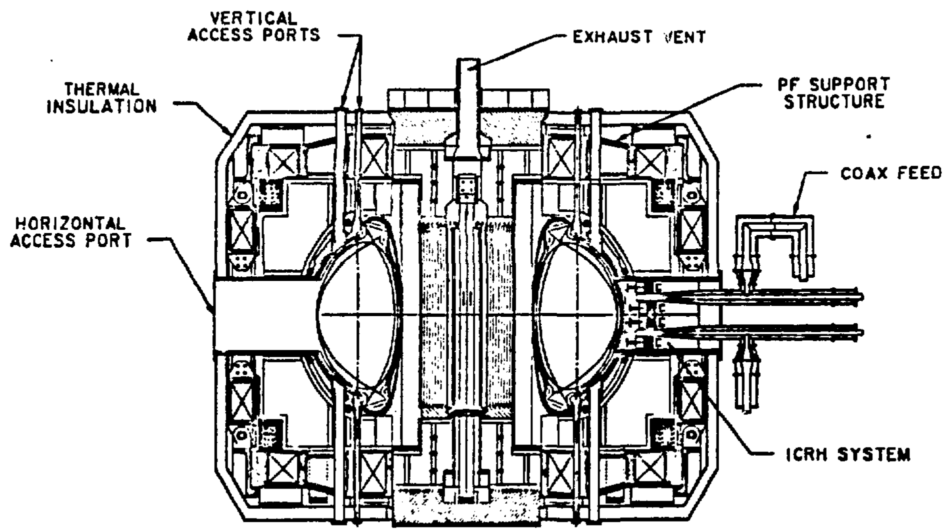

Fig. 1. Elevation view of CIT.

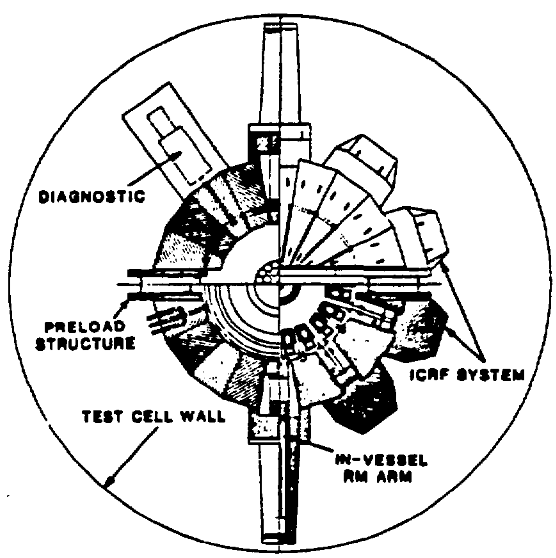

Fig. 2. Plan view of CIT.

2.1. TF Coils

The toroidal magnetic field on axis of 10.3 T is generated by nitrogen-precooled coils that heat adiabatically during a pulse. The inner leg portion of each of the 15 plates of each TF coil is a composite of explosively bonded Inconel to copper, which provides high strength to withstand the large wedge face pressure and still maintain relatively high electrical conductivity. Insulation is provided by molded polyimide glass sheets. There are no internal cooling passages in the plate conductors; cooling between plasma pulses is by edge cooling of the plates by nitrogen.

A The conductor in the nose region is designed as a sandwich with a core of highly conductive copper, CDA-102, at least half-hard and cross-rolled; the facing material is Inconel 718 explosively bonded to the copper. The Inconel is machined to a taper that goes to zero at the end of the nose region. The average current density in the inner leg of the coil is about 7 kA/cm^2. The average temperature at the end of the pulse is about 300 K. The inductive energy deposited in the TF coils during a pulse is about 1.4 GJ.

A coil case and associated intercoil structure are used at the outer TF leg of the TF coil and support the conductors against out-of-plane loads. The external structure also is used to support the vacuum vessel and the outboard PF coils. The coil case serves as a manifold for the liquid nitrogen used to cool the outer leg of the coil.

2.2. PF Coils

The poloidal field magnet system provides the equilibrium, control, and shaping fields for the plasma, as well as the flux change that induces the plasma current and associated ohmic heating. All coils play an active part in fulfilling all of these functions. Two coils comprise the central solenoid, and three pairs of other PF coils are located external to the TF coils as shown in Fig. 1.

The external PF "ring" coils are composed of a double pancake of wide, relatively thick copper conductor, epoxy insulated between turns, and cured after winding. A case surrounds the copper conductors. Liquid nitrogen edge cools the conductors.

The two central solenoid coils are composed of layered pancakes; each pancake is cut from a single laminated plate of Inconel-copper. Stress conditions require a composition of about 50% copper, 40% Inconel, and 10% insulation. Shear loads are taken in pins. The outer boundary of the coil is a grooved insulating layer with channels for liquid nitrogen cooling. The magnetic loads on the solenoid are very large; the maximum fields in the solenoid are about 23 T.

2.3. Vacuum Vessel, First Wall, Limiter, and Divertor

The vacuum vessel is a single-wall toroidal shell of welded and formed Inconel 625. The vessel is prefabricated in two 180° sectors that are placed inside two halves of the TF assemblies and joined together in-situ by welding. Radial and vertical port extensions are then welded into the structure. The radial port extensions have radial keys that engage keyways fastened to the sides of the TF coil cases, which support the vessel, maintain its position, and provide for thermal movements. The ports are closed with double, concentric metallic seals. Poloidal cooling channels welded to the vessel wall are also used to bake out the system.

The first wall consists of POCO AXF-5Q graphite tiles on both the inboard and outboard surfaces. The typical tile is circular, 8 cm in diameter and 1 cm thick, with a central internally threaded foot that is attached to the vessel by screwing it onto studs welded to the wall. Two types of tiles are used in an overlapping pattern to completely cover the surface; the tiles are adjustable in height. The tiles are capable of withstanding the heat loads without contaminating the plasma; they are cooled after each pulse by conduction to the actively cooled vessel. The maximum allowed tile operating temperature is 2500 K.

Top and bottom divertor plates are used to receive most of the plasma energy during divertor operation. The plates are shaped to result in a relatively uniform energy deposition over the plate surface. The divertor plates are composed of many modules, each of which consists of a stainless steel support covered with temperature-resistant POCO AXF-5Q graphite tiles.

2.4. RF Heating

Ion Cyclotron Resonance Heating (ICRH) supplies auxiliary heating power to increase the plasma temperature to ignition conditions. A baseline design has been developed that consists of a pair of resonant double loop antennas in a CIT large port, with one antenna above the other. The maximum power frequency has been selected at 95 MHz; the frequency range at reduced power is 80-110 MHz. A total of 12 MW (at the source, or 9.6 MW to the plasma) is provided. The device can accommodate additional rf power if experimental results show it to be required.

A variety of heating modes are accessible with a single rf source and antenna configuration by choosing the resonant ion species, toroidal field, and wave frequency. The primary scheme selected for CIT is ^{3}He minority at high field (10 T) and H minority at low field (6 T). One advantage of ^{3}He minority heating is the expected strong single-pass absorption even during the start-up phase, which could be replaced by second harmonic tritium heating as ignition conditions are approached.

2.5. Electrical Power, Fueling, and Conventional Facilities

The CIT electrical power system requirements are large and include the systems required to deliver controlled dc power to the various coil systems and the systems to deliver 60-Hz, ac power at appropriate power and voltage levels to the ICRF and facility systems. All of the magnet power conversion systems will be fed from the existing motor-generator-flywheel energy storage system at the Tokamak Fusion Test Reactor (TFTR).

The plasma will be fueled by puffing gas as well as injection of pellets of solid hydrogen isotopes. Pellet injection of both deuterium and tritium is planned for ignition experiments; the velocity required is about 2 km/s, which is about 50% greater than currently achieved velocities.

The CIT has been designed to maximize the use of existing facilities at the PPPL site adjacent to TFTR. A CIT support building will be constructed adjacent to the present TFTR test cell. The CIT test cell will be relatively small, having about 20% of the TFTR test cell floor area, and will be surrounded by decontamination, hot, warm, and tritium injection cells.

3. COSTS

A capital cost estimate has been prepared to determine the new funds required for CIT assuming use of the PPPL site and appropriate equipment and facilities. The cost estimate includes the cost of hardware, engineering, installation, and modifications; the costs include direct and indirect costs. Costs were developed system by system. Contingency costs by system were included. The estimated new funds required for CIT are $285 million based on constant 1986 dollars.

4. SCHEDULE

A total project schedule has been developed based on schedules projected for all major systems. The schedule calls for completion of conceptual design in FY 1987, initiation of detailed design and component fabrication in FY 1988, completion of device installation in FY 1992, and first plasma at the beginning of FY 1993.

5. ACKNOWLEDGMENTS

Major contributions to the CIT program have been made by numerous scientists and engineers from throughout the U.S. fusion program. The design team is composed of individuals from the Princeton Plasma Physics Laboratory, the Fusion Engineering Design Center, Massachusetts Institute of Technology, GA Technologies, Oak Ridge National Laboratory, Idaho National Engineering Laboratory, Los Alamos National Laboratory, and Lawrence Livermore National Laboratory. The project manager is J. A. Schmidt of PPPL. Considerable support and guidance have been provided by the Ignition Technical Oversight Committee chaired by H. P. Furth of PPPL. Thorough consideration of physics aspects associated with CIT has been provided by a national Ignition Physics Study Group chaired by J. Sheffield of ORNL. Programmatic guidance and leadership have been provided by P. M. Stone and T. R. James of the Office of Fusion Energy, U.S. Department of Energy.

6. REFERENCES

[1] COPPI. B., "Physics and Technology of Ignition Experiments," Eleventh Symposium Fusion Engineering, Austin, Texas, November 1985. Also see Ignitor section of "Assessment of Engineering Issues for Compact High Field Ignition Devices," ORNL/FEDC-86/1, Oak Ridge National Laboratory (April 1986).

[2] BROMBERG, L., BOBROV, B., COHN, D. R., KOHSE, G., WILLIAMS, J. E. C., WITT, R., YANG, T-F., LISTVINSKY, G., BERWALD, D., BELL, G., FREISE, S., LUTZ, R., WAGNER, C., MCGRATH, R., "Engineering of a LITE (Long Pulse Ignition Test Experiment) Device," Eleventh Symposium Fusion Engineering, Austin, Texas, November 1985.

[3] BUSHNELL, C. W., "PPPL Ignition Studies Project," Eleventh Symposium Fusion Engineering, Austin, Texas, November 1985.

[4] CITROLO, J. C., "Mechanical Design Aspects of the ISP Device," Eleventh Symposium Fusion Engineering, Austin, Texas, November 1985.

[5] BROWN, T. G., LEE, V. D., MAYHALL, J., "Configuration Development and Structural Assessment of the FEDC Ignitor Concept," Seventh Topical meeting on the Technology of Fusion Energy, Reno, Nevada, June 1986.

[6] FLANAGAN, C. A., ed., "Assessment of Engineering Issues for Compact High Field Ignition Devices," ORNL/FEDC-86/1, Oak Ridge National Laboratory, (April 1986).

[7] BECKER, H., BONANOS, P., CHEN, C. K., DALESSANDRO, J., and MONTGOMERY, D. B., "Common Basis Structural Analysis of Alternate Concepts for the Compact Ignition Experiment," Eleventh Symposium Fusion Engineering, Austin, Texas, November 1985.