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Recent Progress on the Compact Ignition Tokamak (CIT)

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This report describes work done on the Compact Ignition Tokamak (CIT), both at the Princeton Plasma Physics Laboratory (PPPL) and at other fusion laboratories in the United States. The goal of CIT is to reach ignition in a tokamak fusion device in the mid-1990's. Scientific and engineering features of the design are described, as well as projected cost and schedule.

I. Introduction

Ignition is the major near-term goal of fusion research. Studies have been under way in the United States for some time on the possibility of proceeding to ignition within the resources likely to be available to the national program. The conceptual design work has led to a proposal for a compact (R = 1.2 to 1.3 meter), high-field (B = 10 to 11 Tesla) tokamak device capable of achieving ignition and equilibrium burn. The name of the device proposed is the "Compact Ignition Tokamak," or CIT. If sited at the Princeton Plasma Physics Laboratory (PPPL), the project could reuse TFTR equipment, as well as radio-frequency (rf) power supplies now at other US locations. The estimated cost of CIT is about $300M, plus operating expenses. At the close of FY86, the federal administration was evaluating the CIT Conceptual Design Report with a view to project authorization in FY88.

In the CIT design, a hydraulic press reacts the vertical separating force on the toroidal-field (TF) coils. The TF and poloidal-field (PF) conductors use explosively bonded laminates of steel and copper to attain the necessary strength and conductivity in the compact configuration. Ion cyclotron heating in the range 80 to 110 MHz will work on ^{3}He-minority and/or tritium-second-harmonic resonances for the full-field pulses, and hydrogen and deuterium resonances during reduced-field trials. The design features a plasma elongation in the range of 2.0 and a double-null poloidal divertor. The plasma current is 9-10 MA and the equilibrium burn time is about 3 seconds — roughly $10\tau_E$.

Most of the major US fusion research organizations[1] have specific responsibilities in the CIT project: MIT for the PF System, ORNL for radio-frequency supplies and couplers; GA Technologies for the vacuum vessel, its interior hardware and its remote maintenance; Los Alamos for tritium systems and fueling; Livermore for instrumentation and control; PPPL for the TF system, electrical power, diagnostics, cryogenics and cooling. The FEDC in Oak ridge is responsible for design integration and external structure design and the INEL for safety and environmental analysis, as well as conventional facilities.

II. Historical Background

Until the end of 1983, proposals focused on a facility that cost more than one billion dollars and offered the potential to be nearly prototypical of a fusion reactor. The device would ignite, that is, conditions for the production of fusion power would be maintained by self-heating from the alpha particles of deuterium-tritium fusion. Additionally, experimental times would approximate the resistive time scale in the plasma, several hundred seconds, and operation would test significant engineering features of a reactor, for example super-conducting toroidal-field coils.

Around 1984, the fusion community began to study the option of achieving a burning plasma in a minimum-cost device, while economizing on aspects that will be desirable for the long-term development of fusion power. This led to the consideration of a range of parameters similar to those embodied in the "IGNITOR" (Ignited Torus) concept.[2]

In 1985, four reasonably complete conceptual designs were offered for review in national and international forums. The designs were:

1) The Ignition Studies Project (ISP) of PPPL,

2) The Long-pulse Ignited Tokamak Experiment (LITE) of MIT,

3) The version of IGNITOR by the FEDC, and

4) The current version of IGNITOR developed by B. Coppi and associates.

By October 1985, the process of selecting a concept for further development was under way. It was followed by the creation of a national organization responsible for developing one concept into a proposal for DOE funding in the federal budget of fiscal year 1988.

III. Basic Decisions

A. Technical Concept

The nature of the four conceptual designs can be summarized as follows. All concepts have coils that are cooled to liquid nitrogen temperature prior to operation. No heat is removed from the coils during the experimental time (about ten seconds) during which the temperature of the coil rises to approximately room temperature. While the mechanical stresses in each design exceed the nominal allowable values in common use in the engineering of civil and industrial hardware, the stresses do not the exceed values selected by a panel of experst for this particular application. The designs differed in the method used to react the large magnetic forces.

The toroidal-field coils of LITE are self-supporting through use of a high-strength alloy of beryllium and copper. The ISP design incorporates spring-loaded electrical contacts[3] within the TF coil. This allows the inner leg of the TF coil to experience less of the vertical separating force and hence be smaller in cross section. The vertical separating force of the ISP toroidal field coils is reacted into a large external C-clamp. Because the TF coil has a slip joint, the TF coil is demountable, and the vacuum vessel can be installed in one piece. This leads to possibilities for a more robust construction of the vacuum vessel than in the other concepts. The IGNITOR design uses a steel clamp aided by a hydraulic press.

In IGNITOR, the TF coils are restrained by the combined action of wedging against one another and bucking on the inner ohmic heating (OH) transformer. This produces lower stress levels than the wedging-only approach used in LITE and ISP.

LITE and IGNITOR have an external PF coil set, meaning that the TF and the PF are not linked. For ISP the PF and TF are linked, but the joint in the TF eliminates assembly problems.

The process of settling on one design approach for further work and for preparation of a proposal involved managers from ISP, LITE, and IGNITOR projects. Guidance was given by the Ignition Technical Oversite Committee (ITOC) and engineering consultants to that committee. Some key points are:

1) Combined bucking and wedging in the TF has difficulties with proper fitting and bearing of the surfaces, therefore leading to indeterminate load paths;

2) The sliding joint, while workable, would require extensive testing to demonstrate reliability;

3) The good plasma behavior commonly associated with a divertor in present-day experiments suggests that a divertor should be incorporated in the next design iteration.

The result of the deliberations was to focus on a design with features of IGNITOR and LITE, but with some additional variants. In particular, a divertor is to be incorporated. Also, TF and PF are to use explosively-bonded laminates of steel and copper, and the hydraulic press is to be added to aid in carrying the vertical separating force on the TF.

B. Administrative Organization

Concurrently with the process of selecting a design concept for CIT, the Department of Energy (DOE) and the fusion laboratories agreed on a national organization, shown in Fig. 1, to carry the selected concept to a state of conceptual design suitable for a proposal to DOE This proposal would be presented to the US Congress for authorization of the CIT in the fiscal year 1988 budget. The Ignition Physics Study Group (IPSG), shown in the organization, represents the community of fusion physicists that gives advice to the CIT project on the suitability of scientific analysis supporting the engineering design.

From January to June 1986, the participants in the CIT Project developed a proposal, including a plan for research development and a plan for diagnostics.[4] The following section summarizes the content of the proposal and associated documents.[5]

IV. Conceptual Design

The mission of CIT is to realize, study, and optimize ignited plasmas. Doing this at the lowest reasonable cost means building a device that has a major radius of 1 to 1.5 meters, and a pulse length around 5 to 10 seconds. Criteria were established to make sure that the important parameters of size and time were not reduced without careful consideration. The primary requirements were that there should be an adequate margin for ignition with auxiliary heating for scaling laws currently under discussion; that the burning should last $10\tau_E$ and the TF should be steady for $12\tau_E$; that CIT not exceed certain generally accepted limits on beta[6] (β) and density[7] (n_e); and finally that elongation (κ) and aspect ratio (R/a) be near 2 and 3, respectively. With these requirements, the design summarized in Table 1 was prepared.

Table 1: Some important parameters of the CIT design.

PARAMETER (UNITS)	VALUE
Major Radius (meter)	1.22
Minor Radius (meter)	0.45
Elongation (κ)	1.80
Plasma Current, Limiter (MA)	10
Plasma Current, Divertor	9
Toroidal Field (Tesla)	10.4
Time of Steady Field (sec)	3.7
ICRH Initial Power (MW)	10
ICRH Possible Power (MW)	20
Number of Full Field Pulses	3000
Number of Half Power Pulses	50,000
Fusion Power (MW)	300
Burn Time, (sec)	3.1
Divertor Heat Flux, (MW/m^2)	9.5

A. Confinement

Ignition in CIT requires a confinement time of 0.2 - 0.4 seconds, depending on details of the density and temperature profiles. Energy confinement in tokamaks heated with power auxiliary to ohmic heating is commonly discussed in terms of two classes, "L-Mode" and "H-Mode" — for *L*ow and *H*igh confinement.[8] However, these two modes have several different parameterizations in the literature. As a result of differences in profiles and differences in confinement models, predictions from computer modeling cover ranges that are quite broad. The range of ignition is compared to the range of predictions in Fig. 2. The top two bars represent predictions of τ_E from H-mode scalings, to be compared with the range of requirements in the bottom (shaded) bar. One can conclude that, generally, it is possible to gain ignition in CIT, but that there are combinations of models and profiles that will not reach ignition. The remaining unshaded bars show the range of predictions from L-modes. Ignition is possible with L-modes, but somewhat more problematic.

Some results of a time-dependent simulation for an H-mode case with 9 MA of plasma current (appropriate to operation with a divertor) are shown in Fig. 3. The ramp time for both the toroidal field and the current is 3 seconds, and 20 MW of ion cyclotron resonance heating (ICRF) for 1.5 seconds is sufficient for ignition. The jagged shape of the behavior of the peak electron temperature ($T_e(0)$) and the peak alpha-particle density ($n_\alpha(0)$) is due to the well-known sawtooth phenomenon in tokamaks. The sawtooth is a relaxation oscillation in the center of a tokamak plasma which causes periodic and rapid expulsion of energy from the center and, as a result, tends to resist the temperature and density increases in the center of the plasma required for ignition.

Ignition can be achieved with L-Mode confinement if the sawtooth phenomenon can be delayed.[9] The top portion of Fig. 4 shows the result of transport calculations in which two pellets were injected into an initially cold plasma to cause a strong central peak in the density profile; 20 MW of ICRF was then applied to heat the plasma off-axis, at a position half way between the center and the edge, to delay penetration of the current and the sawtooth relaxation. The lower portion of Fig. 4 shows failure to ignite. The same basic transport calculations were used, but sawtooth activity is present.

B. Divertor

The CIT has a divertor for two reasons: to aid in improving confinement by making an H-mode easier to reach; and to help control the density of impurities and fuel particles. Part (a) of Fig. 5 shows a cross-sectional view of the vacuum vessel with divertor plates. The plates are curved on a contour calculated to distribute the heat as evenly as possible after considering the details of the magnetic field shape and the characteristics of energy flow at the edge of the plasma.

An example of the calculated equilibrium is shown in part (b) of Fig. 5. The Tokamak Simulation Code (TSC) used here[10] also allows study of active feedback shape control that is necessary because of the high elongation and the close proximity of the vacuum vessel to the plasma. Each of five top/bottom symmetric coil pairs are part of one or more coil groups, which are designed to control the total plasma current (OH), as well as the major radius, ellipticity, and triangularity. As a result, the plasma can initially have a small elongation which is increased in a controlled fashion as the plasma current is increased. Alternatively, with feedback, the elongation can remain high during the entire phase of current ramp. Studies of the stability of the magnetic configuration have shown that cases of ideal MHD stability exist.[11]

Heat loads in nominal conditions are high, as can be seen in the table of CIT parameters in the previous section. Of further concern are the heat loads during a disruption, in which both magnetic and kinetic energies appear on the wall of the vessel in a short time. The diagram in Fig. 6 shows one approach to estimating the deposition of energy in a disruption of a plasma at or near an ignited condition. Thermal energy of 36 MJ is added to 42 MJ of the 104 MJ total energy in the poloidal magnetic field, causing 63 MJ to be distributed on a relatively slow time scale to the first wall and the divertor (or the limiter, in case the divertor is not operating), and an additional 15 MJ electron thermal quench lasting 1 millisecond is distributed to the divertor or limiter. The diagram notes the various power and energy fluxes.

A special arrangement of graphite tiles covers the first wall, as shown in Fig. 7. A somewhat similar construction is envisaged for the divertor plates, but it is not shown.

C. Heating

Ignition temperatures are achieved through fast-wave ion-cyclotron heating, which has demonstrated efficient ion heating at high power, and can be implemented in a high-density tokamak plasma using sources presently available at reasonable cost. The primary heating method has been chosen to be minority ^{3}He and second-harmonic tritium, with a nominal resonant frequency of 90 MHz for full-field operation. To give some flexibility for operating at lower field, and with minority ^{1}H and second-harmonic deuterium at considerably reduced field, the power system is specified to cover the range 80-110 MHz, with adjustments requiring a few hours. The alternative choice for heating mode, minority ^{1}H and second harmonic deuterium, was considered for use in CIT, but has been deemphasized because of the relative difficulties of the higher frequency and because of possible undesired involvement of alpha particles in power absorption.

The single-pass absorptivity of this mode has been calculated from a one-dimensional mode conversion model[12] based on the assumption that wave absorption is dominated by plasma conditions within a small focal spot in the plasma core. Such calculations, illustrated in Fig. 8, are useful in defining the ranges of appropriate minority concentration and parallel wave number. Minority concentrations are measured by the ratio of minority ion density to electron density, and are 5 % hydrogen in the top D-T case, and 5 % helium-3 in the lower D-T-^{3}He case. These calculations indicate that absorption is adequately strong over a broad range of parallel wave numbers.

Antenna design requires a trade-off of the conflicting requirements to maximize rf coupling and minimize the damage to the antenna from heat flux from the plasma, which results in significant impurity generation. An antenna array based on presently existing inductive loop technology appears to be workable in the CIT design, provided the loops are adequately protected in recesses in the vacuum wall. The design employs a double resonant loop structure which is protected by a gas-cooled Faraday shield, as shown in a side view in Fig. 9. Estimated loading values of 3-10 Ω are marginally adequate to allow 3.5 MW of rf power per port (6 ports for 20 MW) as necessitated by space requirements on the device. This value of power density (2 kW/cm^2) exceeds slightly that obtained in present experiments and is the most ambitious feature of the launcher.

D. Mechanical Design

The toroidal magnetic field is generated by liquid nitrogen pre-cooled coils which undergo adiabatic temperature rise during a pulse. The coils are of modified Bitter coil construction, with the turns cut from plate. The inner leg section of each plate is a composite of copper explosively bonded to Inconel. This permits the turns to withstand the high wedge pressures, while still maintaining relatively high electrical conductivity. The turns are insulated with molded polyimide glass sheets. A partial coil case is provided, supporting the outer and horizontal leg sections for in-plane loading. A tight fit between the conductor and the case is economically obtained by the use of inflatable shims. Stainless steel pillow shims inflated under pressure with silica-epoxy transmit the loads from the turns to the coil case. The coils are edge-cooled with liquid nitrogen between pulses.

A case is required to support the magnet conductors against the out-of-plane loads — over-turning moments from the poloidal field interacting with current in the TF coils. The structure supports the vacuum vessel against both gravity and transient loads coming from a disruption of the plasma current. At the interface between sectors of the case, splined teeth are used to transmit the high shear forces between coils. The TF coil and case assembly are illustrated in Fig. 10.

The axial separating force in the center legs of the TF coils is reacted by an external preloading system, which also preloads the OH solenoid. A total preload of about 250 million newtons (56 million pounds) is applied to the top and bottom of the tokamak by force reacted through the external preload "picture frame" structure shown in Fig. 11. The dynamic pressure is provided by a hydraulic assembly located between the top of the tokamak and the external structure. The hydraulic system is designed to reduce the pressure at the end of the pulse to limit the maximum load as the TF coils grow vertically when their temperature rises during the latter portion of the pulse. Consideration is also being given to a fully programmed hydraulic system, in which the pressures follow directly the magnetic pressures.

E. Costs and Schedule

An integrated view of the CIT experiment is shown in elevation in Fig. 12. Figure 13 shows the location of the CIT relative to TFTR and other landmarks, assuming that PPPL is the site eventually chosen. Using that assumption, the cost estimate summarized in the conceptual design report is $285 million in 1986 dollars, including contingency, but not including operating costs of the experiment, research and development (R&D), and plasma diagnostics. The primary milestones of the construction period are: start construction in October 1987, start assembly in April 1990, and obtain plasma in September 1992. The planned R&D costs about $30 million, and focuses on remote maintenance, shielding, magnet design, and the vacuum vessel combined with the first wall. Diagnostics need funding of $24 million through the end of the construction period.

The conceptual design and associated costs were reviewed by DOE groups whose responsibility it is to assure that projects proposed for line-item funding are on a sound basis. Independent cost estimates did validate the CIT project analysis in total, although there were differences from subsystem to subsystem. Reviews of the technical merits of CIT were also positive. As a result, DOE is seeking line-item funding for the project in the federal budget for fiscal year 1988. The response of the administration will not be known until January 1987.

After the Conceptual Design Report was submitted (June 1986), studies continued on the suitability of all design choices. At the end of FY86, changes in parameters of the reference design were being studied. Some of the more basic changes under consideration are a major radius increase to 1.34 meters and an increase in the number of large ports, from ten to fourteen.

Acknowledgments

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References

[1]The names of institutions cooperating on the Compact Ignition Tokamak have been abbreviated as follows. MIT is the Massachusetts Institute of Technology in Cambridge, Massachusetts. ORNL is the Oak Ridge National Laboratory in Oak Ridge, Tennessee, operated for the Department of Energy by the Martin Marietta Corporation. Los Alamos or LANL is the Los Alamos National Laboratory in Los Alamos, New Mexico, operated for the Department of Energy by the University of California. Livermore or LLNL is the Lawrence Livermore National Laboratory in Livermore, California, operated for the Department of Energy by the University of California. FEDC is the Fusion Energy Design Center, in Oak Ridge. INEL is the Idaho National Engineering Laboratory in Idaho Falls, Idaho, operated for the Department of Energy by the EG&G Idaho, Inc. GA Technologies, Inc., (formerly General Atomic) is a private company operating a major fusion experiment, the Doublet III-D, with funding from the Department of Energy. GAT is located in San Diego, California.

[2]The most recent document describing IGNITOR is: B. Coppi, A. Airoldi, G. Ambrosino, A. Angelini, K. Benz, E. Bittoni, G. Bonizzoni, R. Brunner, G. Burchi, G. Celentano, G. Cenacchi, P. Colestock, A. DeLuca, C. DiNicola, C. Ferro, A. Fubini, F. Garofalo, M. Giannini, K. Greve, M. Haegi, J. Kaelnne, A. Koch, F. Koenig, L. Lanzaveccia, I. Lehrman, E. Mazzucato, P. Moreau, M. Nassi, R. Nicoletta, P. Oliva, E. Pedretti, J. Rauch, C. Rizzello, G. Rubinacci, W. Schminke, D. Scovenna, D. Strickler, R. Terry, P. Titus, G. Volta, "Ignitor Project Feasibility Study," ENEA Contract No. 25878, March 1986, 483 pages.

[3]Holt Murray, "High Current Density, Cryogenically Cooled, Sliding Electrical Joint Development," Princetion University Plasma Physics Laboratory Report PPPL-2370 (September 1986) 42 pp., published in the *Proceedings of the Thirty Second Meeting of the IEEE Holm Conference on Electric Contact Phenomena*, Boston, Massachusetts, October 27-29, 1986, pp. 109-120.

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[12]P. L. Colestock and R. J. Kashuba, "The Theory of Mode Conversion and Wave Damping Near the Ion Cyclotron Frequency," Nuclear Fusion **23**, 763-780 (1983).

Figures

FIG. 1. Organization chart of the Compact Ignition Tokamak. The national nature of the project is indicated by the presence of many widely separated institutions. (86P0245)

FIG. 2. The range in predicted energy confinement times for CIT using a variety of scalings. The length or each box represents the range of predictions using the labeled class of scalings. (ORNL DWG 86 2193 FED, 86P1072)

FIG. 3. Time evolutions of the peak and average temperatures, electron and alpha densities, and toroidal beta . These calculations assume "H-Mode" confinement for a 9 MA plasma operating with a divertor. (86P0080)

FIG. 4. Time evolution of the central ion temperature for two discharges which are programmed differently, but are assumed to have "L-Mode" confinement. The top curve, which achieves ignition, is created by assuming early large rises in the density and strong off-axis heating to keep the current from penetrating. (86P0140)

FIG. 5. Cross-sectional view of the vacuum vessel with divertor plates and plasma boundary (part (a); 86P1038); Calculated equilibrium in which the vessel and optional aluminum plates for passive stabilization are modeled as square ring conductors. (part (b); 86X0713)

FIG. 6. Estimated energy balance during a CIT disruption with the assumptions given in the text. The maximum power and energy flux loads to the first wall, limiter, and divertor plates are shown. For a limiter, the dotted box applies.(86P1018))

FIG. 7. First wall tile arrangement. Part (a) is a line drawing of the tiling of the vacuum vessel wall. Overlapping disks of graphite, each attached to a central hub, have notches in the side of the disk to facilitate the use of remote handling equipment for changing tiles. Part (b) is a solid body drawing made by CAD (computer aided design), but without the detail of the notches. (86P1057 and 86P1094)

FIG. 8. Single-pass absorption calculations for T = 10 keV, $n_e = 5 \times 10^{20} m^{-3}$ with second-harmonic tritium and a small hydrogen contamination, top, and minority ^{3}He, bottom. (86X0618)

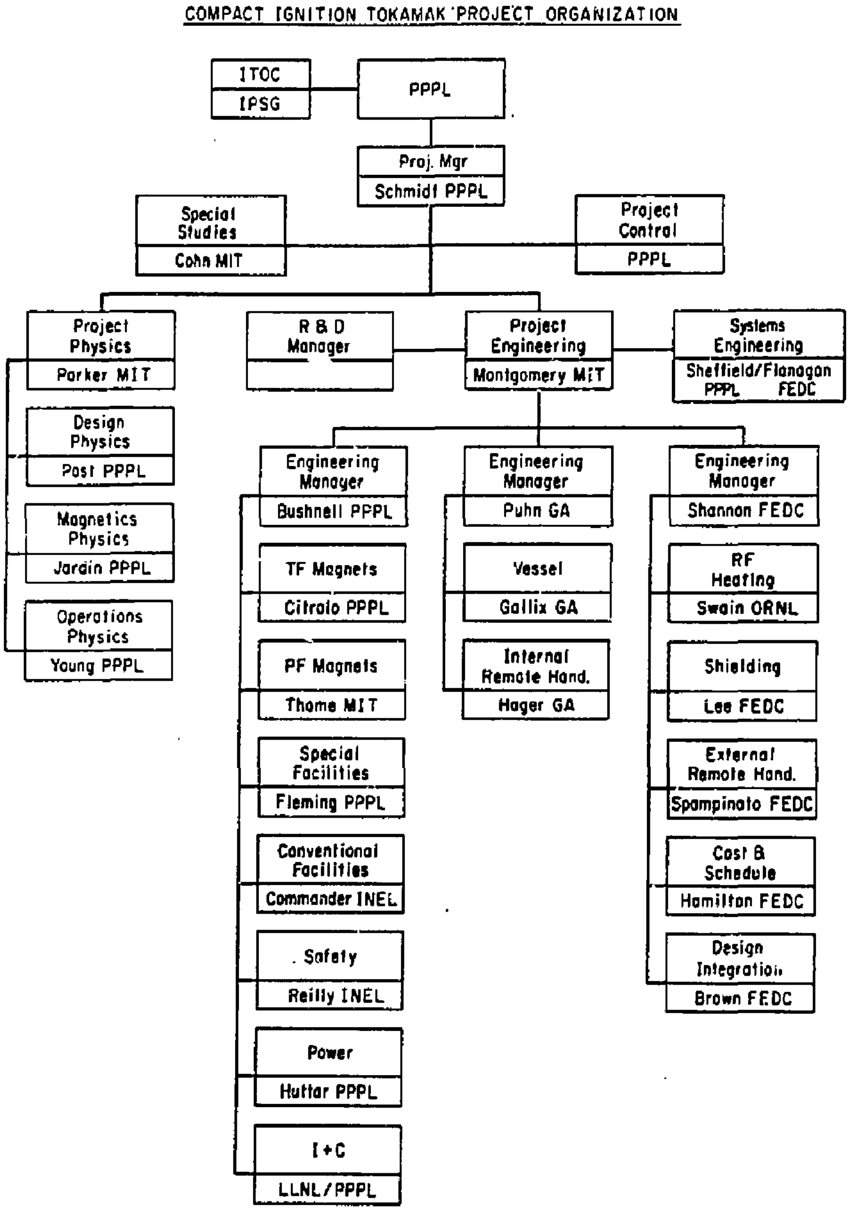
FIG. 9. Side view of the double resonant loop ICRF launcher as mounted in a vessel wall recess. Dual co-axial feed lines enter the vessel through a long and high port, passing between two outboard poloidal-field coils. Power is coupled to the plasma through two resonant double loops, which terminate in capacitors inside the vessel. A Faraday shield prevents improper polarizations from entering the plasma. (86X3278)

FIG. 10. Toroidal-field coil and case assembly. The square teeth are to transmit overturning forces. (86P1032)

FIG. 11. Diagram of the structure which takes the vertical separating force on the toroidal field coil system. The structure is called a picture frame, and is placed in the building before moving in the preassembled CIT tokamak. (86P1026)

FIG. 12. Elevation view of the assembled tokamak, with remote handling equipment, shield, ICRF lines, picture frame clamp, and diagnostic basement. Note that the plane of the picture frame clamp is out of the figure. The close-in proximity shield is approximately a figure of rotation. (86P1040)

FIG. 13. Plan of the intended site at the Princeton Plasma Physics Laboratory which might be used for the CIT. The TFTR buildings adjoin the CIT building at D- Site. Other laboratory and administrative areas at C-Site are to the left – west – of CIT. (86P1056)



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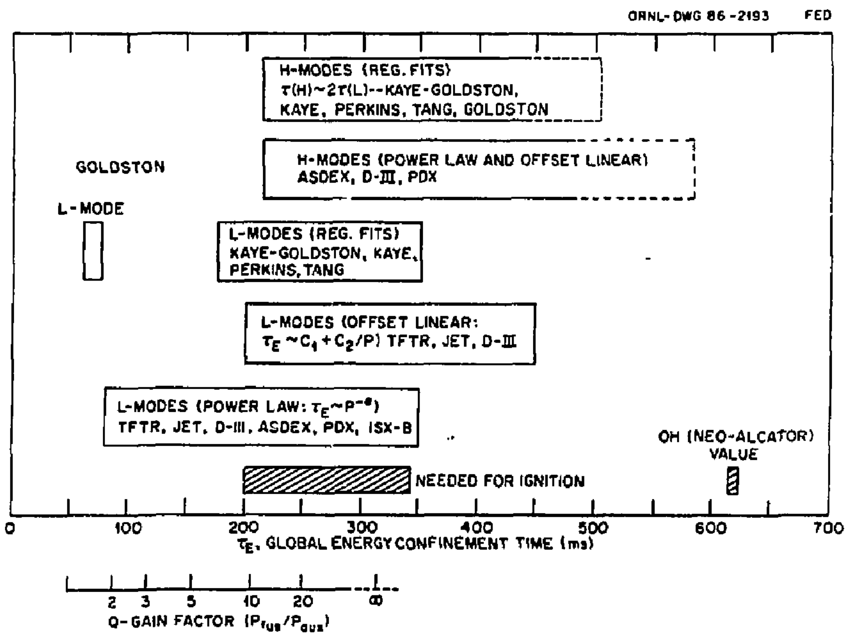

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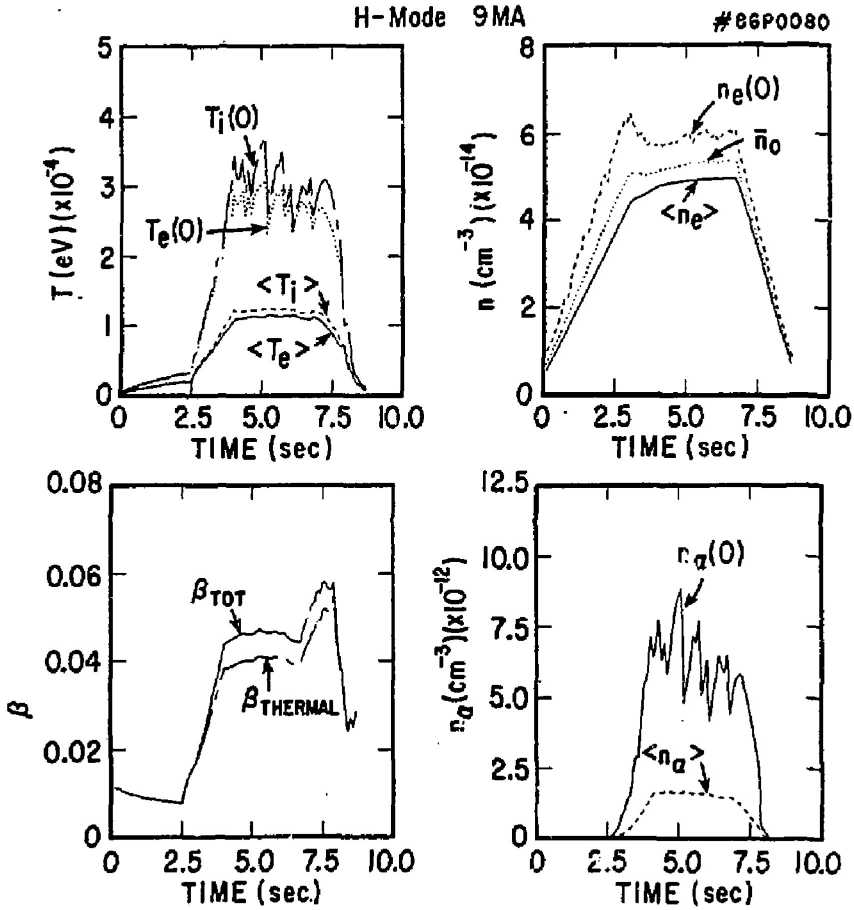

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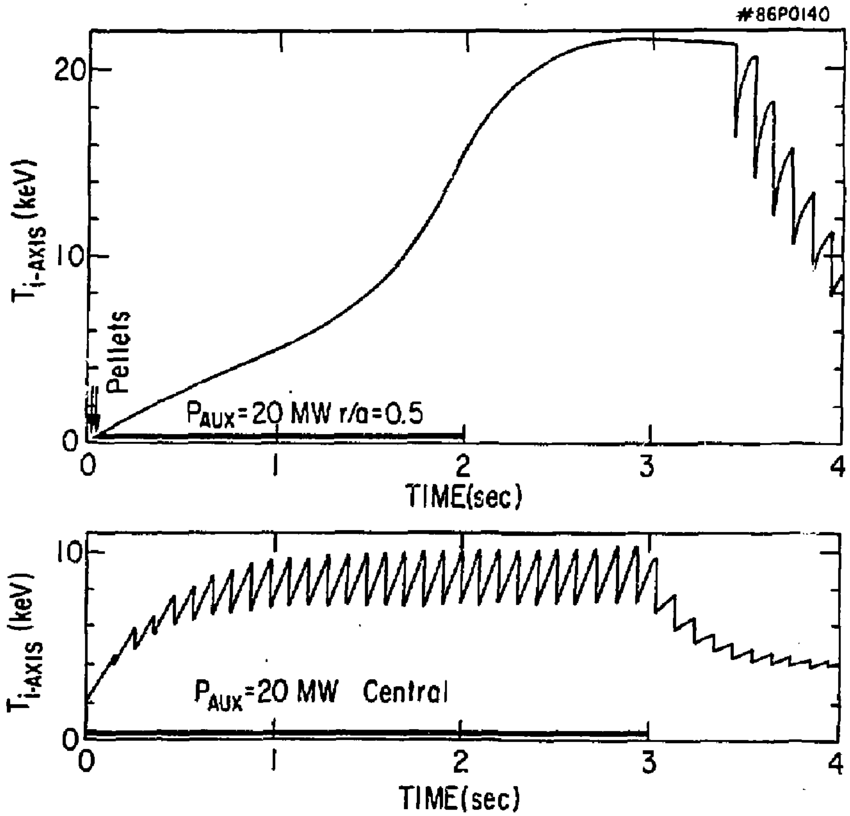

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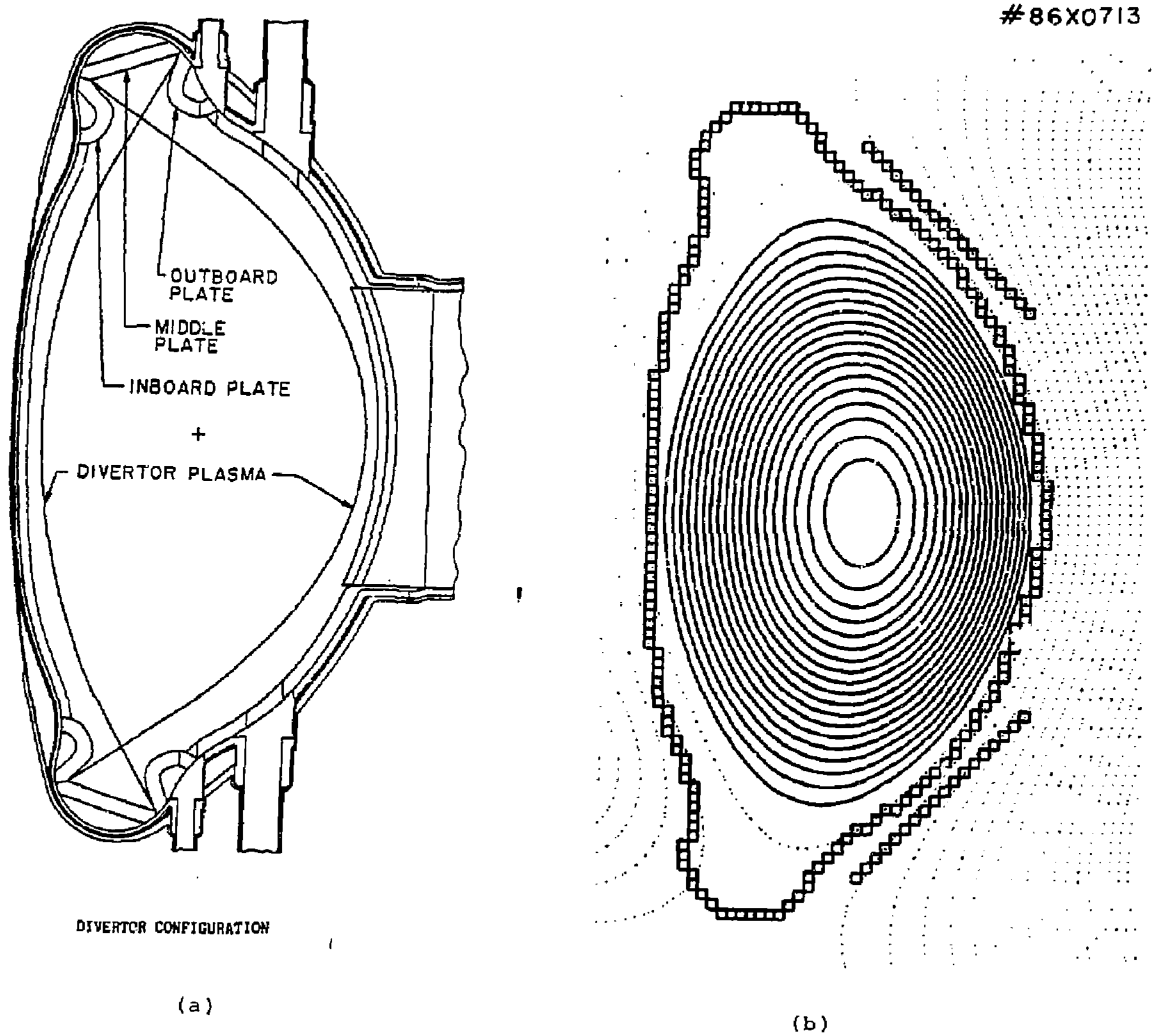

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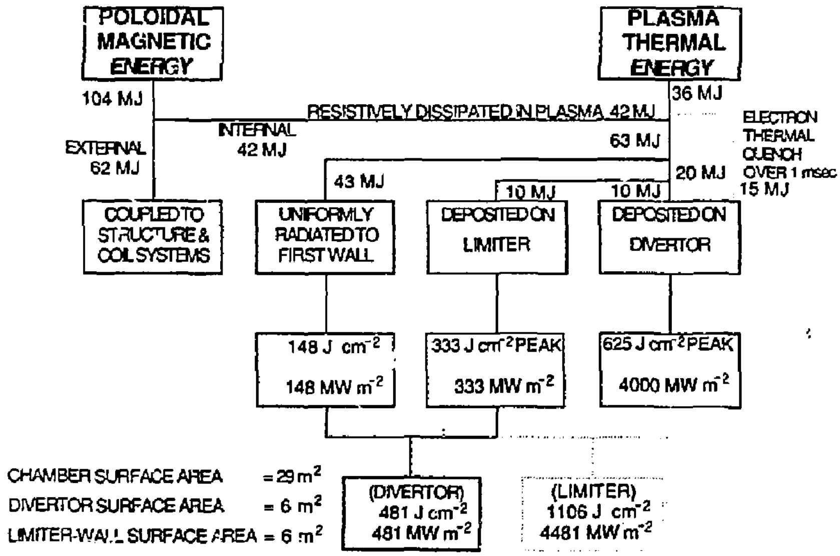

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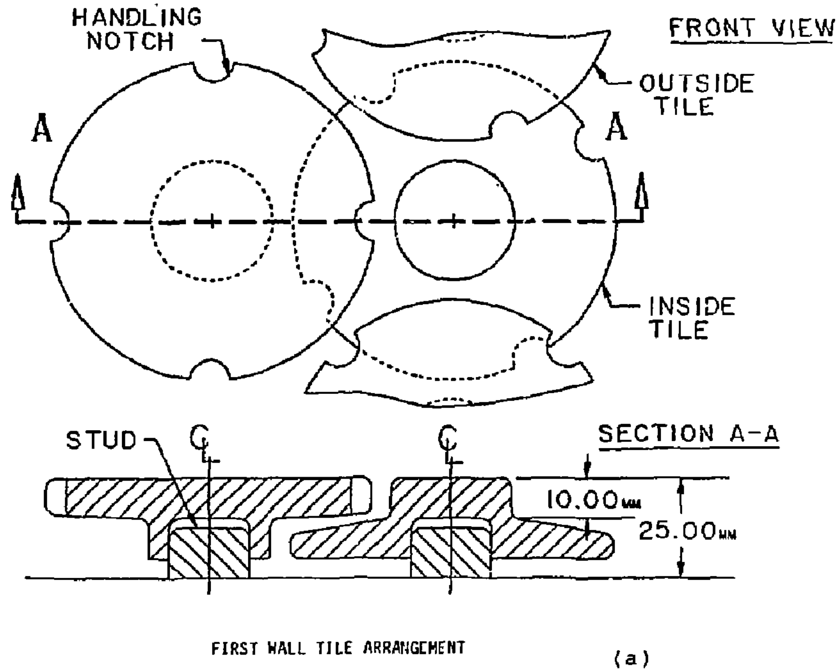
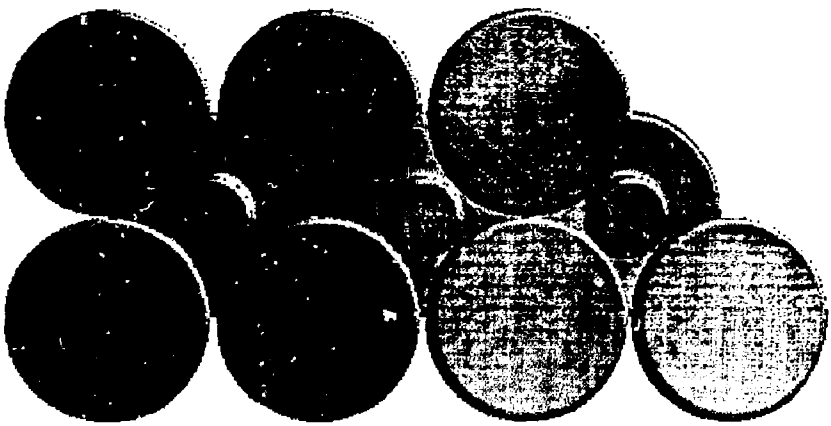


(a)



(b)

FIG. 7. First wall tile arrangement. Part (a) is a line drawing of the tiling of the vacuum vessel wall. Overlapping disks of graphite, each attached to a central hub, have notches in the side of the disk to facilitate the use of remote handling equipment for changing tiles. Part (b) is a solid body drawing made by CAD (computer aided design), but without the detail of the notches. (86P1057 and 86P1094)

#86X0618

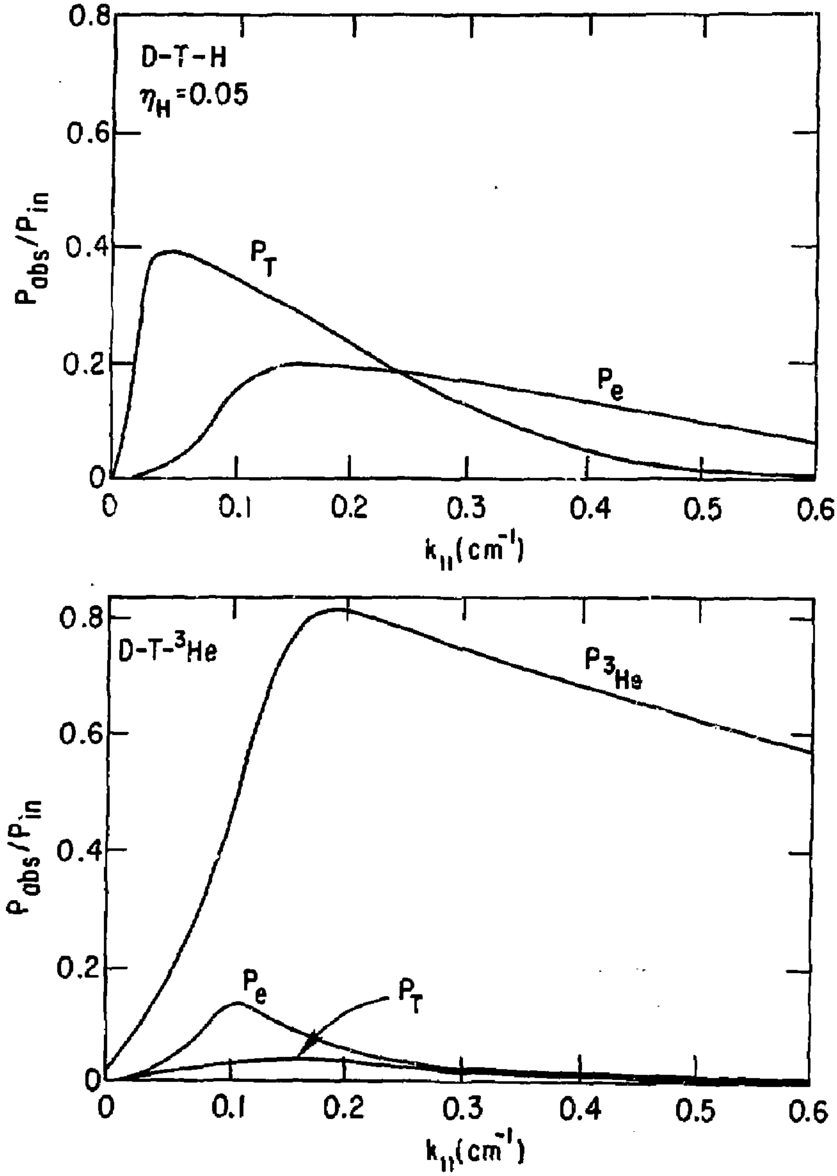


FIG. 8. Single-pass absorption calculations for $T = 10$ keV, $n_e = 5 \times 10^{20} m^{-3}$ with second-harmonic tritium and a small hydrogen contamination, top, and minority ^{3}He, bottom. (86X0618)

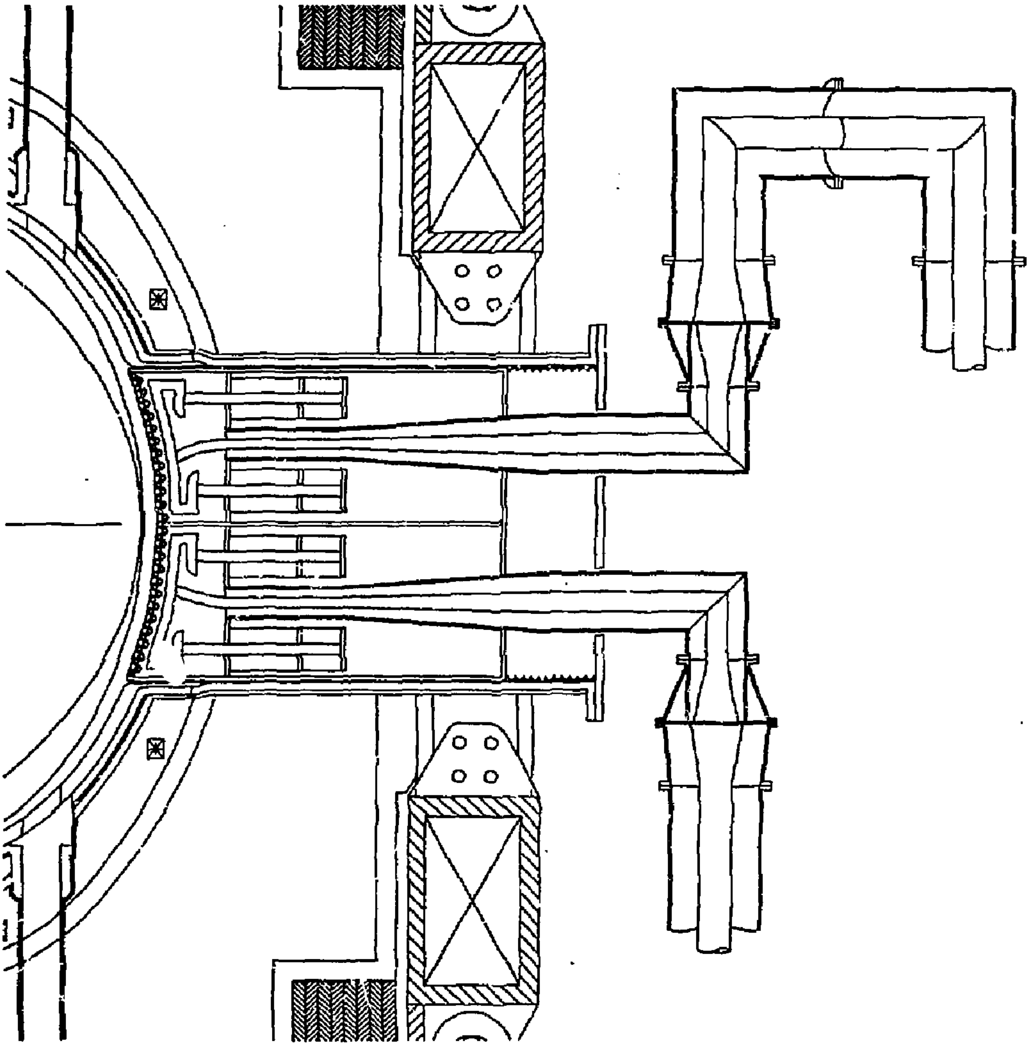


FIG. 9. Side view of the double resonant loop ICRF launcher as mounted in a vessel wall recess. Dual co-axial feed lines enter the vessel through a long and high port, passing between two outboard poloidal-field coils. Power is coupled to the plasma through two resonant double loops, which terminate in capacitors inside the vessel. A Faraday shield prevents improper polarizations from entering the plasma. (86X3278)

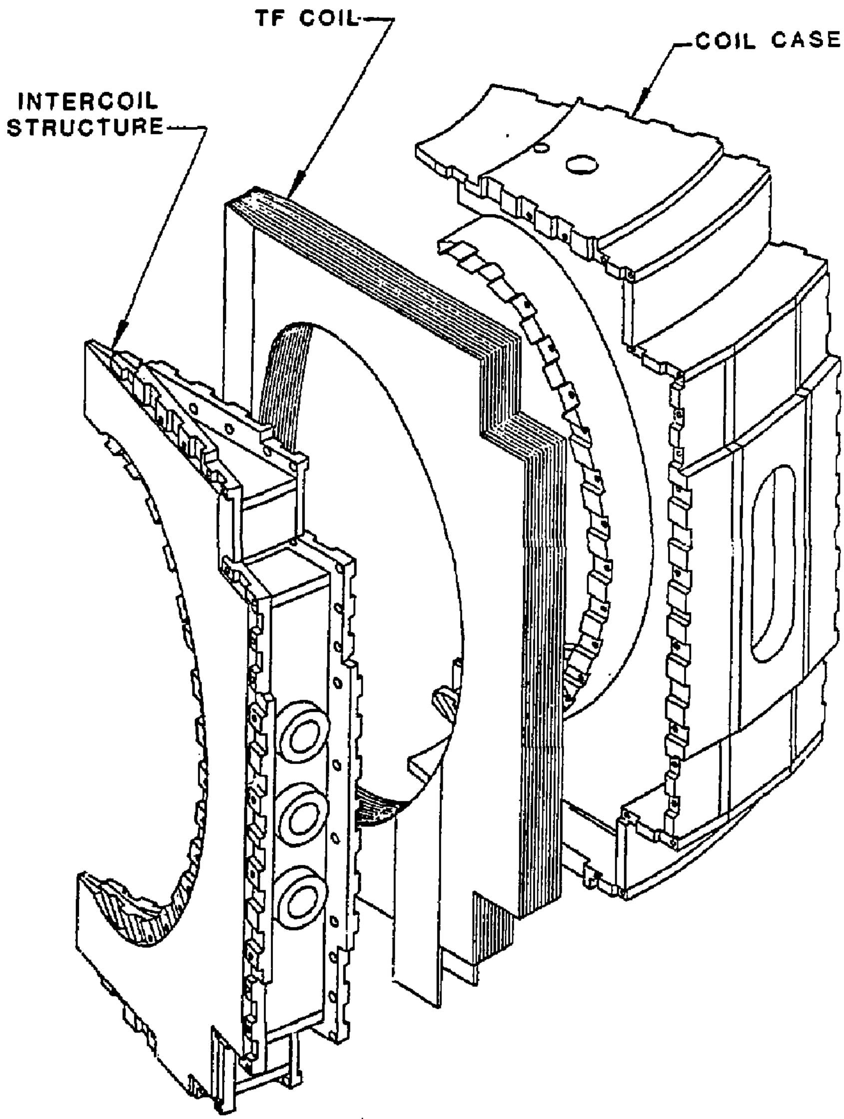


FIG. 10. Toroidal-field coil and case assembly. The square teeth are to transmit overturning forces. (86P1032)

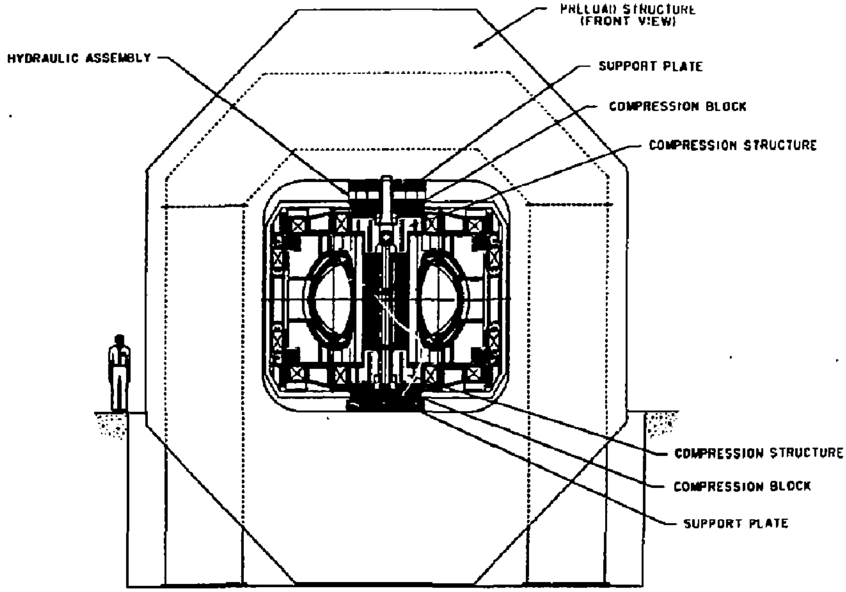
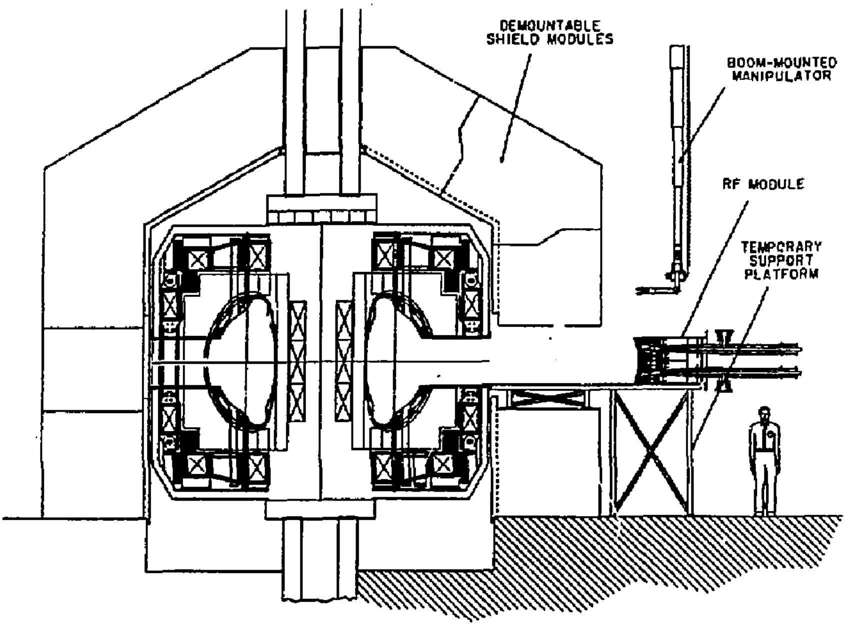
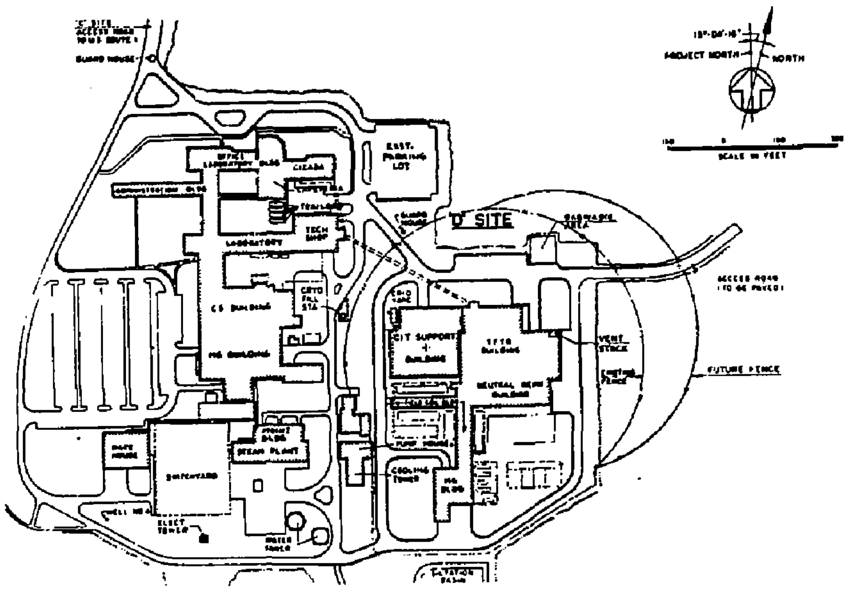


FIG. 11. Diagram of the structure which takes the vertical separating force on the toroidal field coil system. The structure is called a picture frame, and is placed in the building before moving in the preassembled CIT tokamak. (86P1026)



RF MODULE REPAIR IS A TEST CELL OPERATION

FIG. 12. Elevation view of the assembled tokamak, with remote handling equipment, shield, ICRF lines, picture frame clamp, and diagnostic basement. Note that the plane of the picture frame clamp is out of the figure. The close-in proximity shield is approximately a figure of rotation. (86P1040)



CIT FACILITIES SITE PLAN

FIG. 13. Plan of the intended site at the Princeton Plasma Physics Laboratory which might be used for the CIT. The TFTR buildings adjoin the CIT building at D- Site. Other laboratory and administrative areas at C-Site are to the left – west – of CIT. (86P1056)

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