PHYSICS CRITERIA AND DESIGN SOLUTIONS FOR AN ADVANCED IGNITION EXPERIMENT

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The lastest embodiment (Ignitor Ult) of an advanced compact, high magnetic field toroidal machine with the purpose of investigating deuterium–tritium (D–T) fusion ignition conditions is described. The conceptual foundations of the experiment, based on present-day experimental and theoretical understanding of the physics of magnetically confined plasmas, are analyzed. The engineering solutions and design characteristics of the machine's principal components are presented.

1. INTRODUCTION

Since its original proposal [1] in 1975, the Ignitor experiment has undergone a process of continuing development of its plasma parameters [1-5] and its engineering design [6-9]. The goals have remained unchanged:

- to investigate the collective modes and the transport processes that occur in D–T fusion burning plasmas;
- to attain fusion burn conditions and ignition at relatively low peak temperature ($T_{io} \simeq T_{eo} \equiv T_o \lesssim 15$ keV), with values of the confinement parameter $n_o \tau_E \gtrsim 4 \times 10^{20}$ sec/m^3 while avoiding the need or reliance on an injected heating system. (Here n_o is the peak plasma density and τ_E is the energy replacement time);
- to study the effectiveness of the injected heating system in accelerating the approach to burn conditions, in controlling the evolution (central peaking) of the toroidal current density profile, and in stabilizing possible sawtooth oscillations;
- to evaluate the potential of a compact, high field toroidal experiment in order to reach D–^{3}He burning conditions;
- to test diagnostic systems for burning plasmas;
- to develop methods for control, heating and fueling of high density plasmas.

2. PHYSICS BASIS

The Ignitor experiment was conceived on the basis of the well known properties of high density plasmas, in terms of good confinement and high degree of purity [10]. The design parameters of Ignitor Ult are reported in Table I. These parameters have been chosen in order to obtain:

- a high peak plasma density ($n_o \simeq 10^{21}$ m^{-3});
- a high mean poloidal magnetic field and a large toroidal plasma current;
- low poloidal betas ($\beta_p \lesssim 0.15$ at ignition);
- a small volume of the region where $q < 1$.

TABLE I: Ignitor Ult Design Parameters

$R_o \simeq 1.32$ m	Major radius
$a \simeq 0.47$ m	Minor radii
$\kappa \simeq 1.85$	Elongation
$\delta_G \simeq 0.4$	Triangularity
$I_p \lesssim 12$ MA	Toroidal plasma current
$I_\theta \lesssim 9$ MA	Poloidal plasma current
$B_T \lesssim 13$ T	Vacuum toroidal field at R_o
$\Delta B_T \lesssim 1.4$ T	Paramagnetic field
$\langle J_\phi \rangle \lesssim 9.3$ MA/m^2	Volume-average toroidal current density
$\bar{B}_p \lesssim 3.75$ T	Mean poloidal field
$I_p \bar{B}_p \lesssim 45$ MN/m	Confinement strength parameter
$q_\psi \simeq 3.5$	Edge safety factor for $I_p \simeq 12$ MA
$S_o \simeq 36$ m^2	Plasma surface area
$t_r \simeq 3$ to 4 sec	Ramp up time for I_p
$t_{ft} \simeq 4$ sec	Flat top time at maximum B_T
$P_J \lesssim 16$ MW	Injected heating power (ICRH)

These characteristics should allow the achievement of:

- a strong rate of ohmic heating up to ignition. This is accomplished by programming the initial rise of I_p and n_o while gradually increasing the cross section of the plasma column. Thus, the electric field is strongly inhomogeneous and small at the center of the plasma column where the temperature can reach relatively high values, and is maximum at the edge of it (corresponding to loop voltages $V_\phi \sim 1$ V).
- ignition at low temperature, $T_o \lesssim 15$ keV.
- limitation of the degradation in τ_E that has been observed so far when a form of injected heating is applied at discrete points around the torus, with a power much larger than that of ohmic heating. Our strategy is to sustain a strong rate of ohmic heating up to relatively high temperatures, where fusion α–particle heating becomes significant. A comparable degradation of τ_E due to the α–heating is by no means certain as this form of heating is internal to the plasma and distributed axi-

symmetrically, two features that it shares with ohmic heating that has optimal confinement characteristics.

- a high degree of purity that prevents dilution of the reacting nuclei and loss of internal energy from the plasma core by radiation. Experiments have confirmed that Z_{eff} is a monotonically decreasing function of the plasma density and is close to unity when $n_o \simeq 10^{21}$ m^{-3}. The high values of B_T and the relatively low thermal loads on the first wall, which are expected in Ignitor under low temperature ignition conditions, are further favorable factors to obtain low values of Z_{eff}.
- a relatively high plasma edge density that helps to confine impurities to the scrape off layer, where the induced radiation helps to distribute the thermal wall loading more uniformly on the first wall;
- peaked plasma density profiles (maintained by external means such as a pellet injector, if necessary) that ensure stability against the so-called η_i modes driven by the ion temperature gradient.
- good confinement of the thermal plasma and the α-particle population in the central part of the plasma column. The large currents that can be produced confine the α-particles to deposit their energy in the central region, where the diffusion coefficient for the plasma thermal energy is consistently found to be minimal.
- a paramagnetic plasma current I_θ (up to 9 MA), that increases B_T at $R = R_o$ by about 10%.
- the best possible margin for stability against ideal MHD and resistive $m^o = 1$ modes [11], that could hamper the attainment of ignition [12].

2.1 Plasma Performances

Recent results of free boundary numerical simulations [2-5], using the Tokamak Simulation Code [13], have shown that ignition is most effectively achieved soon after the end of the current rise, when the volume of the region where $q < 1$ is less than 1/10 of the total plasma volume. This situation, combined with the low value of β_p that ensures the ideal MHD stability of $m^o = 1$ modes, keeps the effects of potential sawtooth oscillations small.

For the Ignitor parameters given in Table I, ignition can be reached [3], after a 3 sec current ramp, at $t \simeq 4.3$ sec with $T_o \simeq 11$ keV, $n_{eo} \simeq 1.1 \times 10^{21} m^{-3}$ and $n_{eo}/\langle n_e \rangle \simeq 2.2$, where $\langle n_e \rangle$ is the volume average density, corresponding to a thermal stored energy $W \simeq 12$ MJ. If the peaking of the temperature profile $T_o/\langle T \rangle$ is limited to approximately 3, then $\tau_E \simeq 0.66$ sec at ignition, when $Z_{eff} \simeq 1.2$. The α-heating power P_α, that is equal to the total power losses (P_L) at ignition, is about 18 MW, while $P_{OH} \simeq 9.5$ MW. This makes the thermal load on the first wall relatively mild.

During the current ramp, P_{OH} increases continuously. At the same time n_e is also being increased. The maximum ohmic heating power density is generated well away from the magnetic axis (typically, around $r \simeq a/2$), where the temperature is relatively modest. After the current ramp ends, P_{OH} falls gradually as the temperature rises, although not as quickly as the central loop voltage $V_{\phi o} \sim T_o^{-3/2}$. The relatively large values of P_{OH} is due to the fact that, under nonstationary conditions, V_ϕ is a strong function of the plasma radius and is much higher at the edge than in the center.

The flat top phase at $B_T \simeq 13$ T has been designed to last about 4 sec. However, our analyses indicate that the maximum value of I_p is not necessary to sustain the ignited state because the fusion α-power takes care of the power balance. By operating at lower I_p and B_T, after ignition conditions are reached, it is possible to extend the time over which burning conditions can be sustained.

Moderate amounts of auxiliary heating [5], $P_J \sim 5$–10 MW, started during the current ramp, allow ignition with $\tau_E \simeq 0.4$ sec. Similarly [5], in the ohmic case, if the requirement that the $q < 1$ region remain small is dropped, ignition can also occur for $\tau_E \simeq 0.4$ sec.

Given a level of thermal transport and radiation losses (Z_{eff}), there is an optimum density that minimizes the time to reach ignition. In practice, Z_{eff} should not be higher than about 1.6 in the absence of auxiliary heating. A higher density is necessary to compensate for degraded conditions. Better results are obtained by limiting n_o during the current ramp and increasing, if needed, its value during the flat top.

Few megawatts (e.g. 5 MW) of injected heating, started during the current ramp, substantially shorten the time to ignition by increasing the central heating. It also effectively controls the size of the $q \leq 1$ region by raising the temperature in the outer half of the plasma column and slowing the rate of current penetration [3-5]. The high temperatures of the central region (10 to 15 keV) act to "freeze in" the central current density. Thus the $q \leq 1$ region can be kept quite small until well beyond ignition.

3. ENGINEERING CONSIDERATIONS

The main engineering objectives are to:

- create and control the desired variety of plasma confinement configurations;
- induce the toroidal plasma current and maintain the plasma discharge for an adequate number of energy confinement times;
- operate with an acceptable thermal wall loading;
- withstand the static, dynamic, thermal, electromagnetic, and disruptive loads;
- provide access for diagnostics, pellet injector, r.f. antennae, vacuum system, remote maintenance, etc.;
- have a reasonable cooling down time of the toroidal and poloidal field magnet between discharges;
- minimize the electrical power and energy requirements;
- assure adequate reliability and durability of internal and external maintenance systems;

3.1 Poloidal Field System

A highly optimized set of 11 up-down symmetric poloidal field coils (PFCs) [9], placed in the proximity to the plasma column (see Fig. 1), has the function of:

- inducing the plasma current;
- creating the desired plasma equilibrium configurations;
- maintaining them radially and vertically stable.

Oxygen free high conductivity copper has been selected as the material for most of the central solenoid (CS) in order to minimize the energy dissipated ohmically in these coils, while Glidcop material is used for the other poloidal field coils that, in addition, are self-supporting structures. This is possible since these last coils are located in a region of the machine where more space is available and, furthermore, carry high current only during a limited time interval of the discharge. The insulator material is Orlitherm (in-vacuum pressure impregnated epoxy resin and fiber glass).

The CS consists of an array of copper and Glidcop coils wrapped around the central steel pole of the machine. Each conducting coil is provided with a cooling channel at its center (internal diameter = 8 mm). The cooling fluid is gaseous He. The initial temperature before a plasma discharge will be lowered to about 30 K. The final temperature should not exceed 260 K.

The radial outward force acting on the coils of the CS that are close to the midplane is supported in part by bucking against the toroidal magnet coils; the uppermost coil is contained by a stainless steel belt.

3.1.1 Poloidal Flux Requirement. The poloidal flux requirement, for the maximum current scenario, at the plasma edge is $\simeq$ 28 V-sec, corresponding $\simeq$ 33 V-sec at R_o (of which $\simeq$ 12 V-sec are delivered by the vertical equilibrium field coils). This value has been estimated taking into account also the analysis [14] of different plasma scenarios under a range of different values of I_p, ion and electron thermal conductivity, Z_{eff}, n_o, t_r, t_{ft}, etc.

The poloidal field system is designed to produce about 30 V-sec at the plasma edge or about 35 V-sec at R_o.

3.1.2 Plasma Equilibria. Several kinds of plasma equilibria can be produced:

- "limiter" configurations that fill the entire cavity enclosed by the first wall are useful to keep the thermal wall loading as uniformly distributed as possible and to minimize the out-of-plane forces on the toroidal field coils (TFCs);
- "transient (double) x-point" configurations. In this case, it is necessary to keep I_p well below its maximum design value, and to avoid the presence of narrow regions of the first wall where the thermal loading is too high. When the localized thermal wall loading, associated with the x-point configuration, is estimated to exceed the desirable limits, the equilibrium is made to evolve into a limiter configuration.
- "detached limiter" configurations that enable the plasma column to have its outer edge detached from the first wall. This could be necessary to establish the H-regime of enhanced confinement, following a procedure demonstrated by a significant set of experiments [15].

3.1.3 Radial Position Control. The radial feedback system is mainly used to:

- control the plasma current density distribution and the magnetic safety factor evolution during the initial ramp phase of the discharge by changing the dimensions and position of the plasma cross section;
- keep the plasma in the desired position relative to the wall and the Faraday shields of the ICRF antennae;
- control the transition between x-point configurations and the limiter configuration;
- control the radial position during vertical disruption or during the rapid change of plasma parameters, such as the internal inductance (l_i) or the poloidal beta (β_p).

3.1.4 Vertical Position Control. An extensive numerical simulation of the plasma column dynamics has been carried out. The effect of a layer of copper ($\simeq 0.5 - 1.$ mm) deposited over the outer surface of the plasma chamber has been studied. This solution has been considered in order to slow the vertical instability and thus reduce the power requirement and/or allow a longer response time of the controller. A feedback system that uses an outboard coil is capable of controlling the plasma vertical position. By using a combination of one inboard and one outboard coil, the maximum power requirement is reduced further (see Nassi et al., this proceedings).

3.2 Toroidal Field System

In order to make Ignitor suitable for plasma current pulses that can reach and maintain ignition, the TFCs have been designed to operate with lower values of the starting temperature (30 K) and of the current density ($<$ 90 MA/m^2) than in the design of the Alcator-C machine (temperature $\simeq$ 80 K and current density up to 220 MA/m^2).

3.2.1 Structural Solution. The TFCs are made of copper plates, connected in series externally and supported by an appropriate steel structure so as to withstand both the vertical (axial) and the horizontal (radial) electrodynamic forces [9]. In particular, the loads on the inner leg of the TFCs are supported by:

- bucking between the TFCs and the CS (a sliding surface is provided at the interface);
- wedging in the inner part of the toroidal coils;
- external structural elements.

As far as the out-of-plane forces are concerned, two main provisions have been adopted:

- these forces are minimized by reducing the angle between the poloidal magnetic field lines and the current lines within the TFCs. This has been attained by giving the TFCs the same shape as the last closed plasma surface;
- stress concentrations are avoided by using the friction existing between the magnet plates and their supports, instead of keys, bolts or tenons, to take the out-of-plane forces.

3.2.2 Cooling System. A hybrid cryogenic system [16] is adopted for the cooling of the TFCs, where the warmer part (80 K) of the plant is operated with liquid N_2 and the colder part (30 K) with gaseous He. The conductor's temperature before a plasma discharge is assumed to be about 30 K. After a current pulse, corresponding to the maximum magnetic field scenario, the temperature will reach about 230 K in the region facing the transformer (1/3 of its volume) and about 95 K in the remaining part. Heat transfer from the coils to the coolant takes place through forced convection in conduits along the external surface of the coils. The same helium stream passes several times through the magnet and is recooled between successive passages.

3.3 Structural Components

The main mechanical and electrical structural components are [9]:

- a set of 48 steel plates (AISI 316 LN) or "C-clamps" that surrounds each of the 24 modules of the TFCs.

The C–clamps have the following important functions:

- to contain the horizontal expansion forces of the TFCs. This is possible because the C-clamps are embraced at the extremities by the toroidally continuous bracing rings;
- to contain most of the vertical expansion force of the TFCs. This is possible not only because the C–clamps embrace the TFCs, but also because there is a significant vertical pre-compression. In fact, the C–clamps are wedged on the outside to allow the unwedged part to rotate around an effective hinge under the effect of the pressure applied by the bracings rings. In this way the nose of the C-clamps rotate inward, compressing the inner legs of the TFCs. Therefore only a small fraction of the vertical separating force is unloaded onto the central leg of the TFCs;
- to stabilize the machine and give a positive stop to the inward displacement of the C–clamps under the pressure of the bracing rings by means of their wedging. Without this positive stop the bucking between the CS and the TFCs and the wedging of the inner leg of the TFCs could become unknown and excessive;
- to support the other main machine components: plasma chamber, PFCs, etc.;

- two <u>bracing rings</u> that maintain the plate assembly and transfer the vertical separating force produced by the TFCs to the effective outer shell formed by the steel plates. They are continuous, completely laminated and electrically insulated. The two flat terminal faces of each ring are also insulated, since a considerable voltage difference builds up between the two extremities of the laminated sheet.
- a <u>tensioning system</u> composed of a set of 96 rams (48 on the top and 48 on the bottom) acting at both extremities of each of the 48 C–clamps. The rams insert a doubly wedged rod between two wedge female members which are thereby expanded between the C–clamps and the bracing rings. By reversing the movement, the less acute wedge unlocks the expanded members. Each ram is intended to act independently since perfect tolerances in construction and the reproducibility of the local friction factors cannot be foreseen;
- a <u>central post</u> (a massive, electrically insulated structure) that fills the bore of the central solenoid (external diameter = 0.43 m, height = 6.8 m). Radial vertical cuts are made in the post to reduce the induced currents. Its main functions are to absorb the centripetal force acting on the inner leg of the TFCs and to be a component of a central press. It is provided with fir tree root attachments at both extremities for the post heads;
- a <u>pressure piston</u>, that is, a segmented cylinder, electrically insulated, that transfers the force of the electromagnetic press without producing any bending moment on the inner leg of the TFCs;
- a vertical <u>electromagnetic press</u> (30 to 40 MN) that is connected to the central post and is capable of applying a compression preload on the inner leg of the toroidal magnet to reduce the electromagnetic load. The press is deactivated as soon as the thermal expansion due to the temperature rise in the TFCs becomes significant, or whenever the machine is operated with magnetic fields below the maximum considered values;
- a set of <u>supporting legs</u>, attached to the C–clamps, that withstand the machine weight.

3.4 Plasma Chamber

The plasma chamber [9] is made of Inconel 625, is 17 mm thick in the inboard side between the thicker welding ribs, and 26 mm in the outboard side. It is divided into 12 sectors that can be assembled and joined by welding. The chamber is mechanically supported and restrained by the C–clamps by means of vertical, horizontal and transverse supports attached to the long horizontal port ducts. These supports allow for freedom of deformation under electromagnetic and thermal loads. The main consideration in the design of the plasma chamber and its mechanical supports has been the ability to withstand both vertical and axisymmetric disruptions with plasma current decay rates from 1 to 2.5 MA/msec.

The plasma chamber acts as a support for the first wall system. It provides vertical and equatorial access ports for the plasma diagnostics systems, the vacuum system, the pellet injector, the auxiliary heating system, the in-vessel remote maintenance system, etc.

The temperature of the plasma chamber is decoupled from that of the TFCs by adopting a separating insulation layer and independent cooling. Furthermore, the chamber wall can also be independently heated by induction using a 50 Hz power supply for the TFCs. An additional external resistive heating is provided to heat the port regions.

3.5 First Wall System

The first wall [9] is made of graphite tiles (20 mm thick) that cover the entire inner surface of the plasma chamber. Thus, in principle, it works as a completely extended limiter. The major advantage of this solution is the large contact area on which the heat load may be spread. This area is about the total first wall surface ($\simeq 36$ m^2), for plasma limiter configurations filling the entire cavity.

The tiles, are brazed on appropriate (double curvature) support plates of Inconel 625 that can be replaced by a remote handling system. Graphite reinforced by carbon fiber has been selected as first wall material, taking into account both thermo-mechanical and plasma-related properties. The major consideration that has led to this choice of material is the expected effect of disruptions.

No direct cooling system is provided; cooling takes place primarily by conduction to the plasma chamber and by radiation. The necessary cleaning of the first wall is obtained by pulsed discharge, glow discharge and baking at an appropriate temperature.

For a limiter configuration that fills the entire plasma chamber, we may estimate the maximum thermal wall loading to be $\simeq 0.9$ MW/m^2 for $P_L \simeq 18$ MW. On the other hand, if we assume that the plasma, after achieving ignition, will reach a state at $T_o \lesssim 15$ keV where $P_L \simeq 40$ MW, the corresponding wall loading is $\simeq 2.0$ MW/m^2.

3.6 Auxiliary Systems

3.6.1 Injected Heating System. Since ignition can be attained by ohmic heating alone, injected heating systems have the role of backups; to be available, if needed, to suppress the possible onset of sawtooth oscillations, to control the plasma temperature evolution and the current density profiles, and to accelerate the attainment of ignition.

An ICRF system with a frequency $f \simeq 130$ MHz and a maximum power delivered to the plasma $P_J \simeq 16$ MW, has been adopted because of the experimental evidence of its effectiveness in relatively high density plasmas. The antennae are placed in six housings inserted into the plasma chamber and connected to the large horizontal ports.

3.6.2 Pellet Injector. An injector of D or D-T pellets ($\simeq 4$ mm diameter) is considered, in addition to the well-tested technique of gas injection, to create and maintain the desired density profile. Pellet velocities of 2 km/s or higher are required to reach the central region of the plasma column. Another use for a pellet injector is to condition the first wall by launching lithium pellets into the plasma column prior to regular hydrogenic discharges [17].

3.6.3 Tritium Storage and Delivery System A complete study [18] of the tritium storage and delivery system has been concluded by a joint team from the IFP-CNR of Milan and the Ontario Hydro Research Division.

A maximum number of 6 tritium discharges per day has been planned. During each discharge $\simeq 8.7$ mbar l of tritium (corresponding to about 22 Ci) are injected for the prefill and $\simeq 8.7$ mbar l of tritium are injected for the gas filling phase. There are two injection points and the duration of the gas filling process is $\simeq 0.1$ sec.

3.7 Neutron Activation

A study [19] of the machine activation has shown that:

- this has a major role in the operation and maintenance of the machine;
- the short–term activation of the plasma chamber in particular is high, and requires the use of a remote handling system for the maintenance of the inner components soon after the beginning of the planned D–T operation.
- the long–term activation of the plasma chamber material (Inconel 625) is such that it can be classified as low level waste. All the other components are predicted to have low long–term activation and can be recycled, if required;
- the short and long–term activation of the first wall is always negligible.

4. CONCLUSIONS

The set of features that make a high field, tight aspect ratio toroidal plasma confinement configuration suitable for a D-T ignition experiment have been discussed. Achieving relatively high plasma densities, and making the most effective use of ohmic heating, has been shown to be the simplest and most reliable approach to ignition. The time evolution of the plasma toward ignition has been seen to play a significant role in characterizing the ignition requirements. That is, the transient processes produced during the initial phase when the plasma current is ramped to its final value affect the plasma stability and heating at ignition. In particular, compact high field experiments are shown to be suitable to produce low temperature ignition ($T_o \gtrsim 11$ keV) of D-T plasma mixtures with $n_o \simeq 10^{21}$ m^{-3}, under known criteria of both energetics and stability.

The design of a machine capable of complying with its physics requirements has been carried out using the high field magnet technology, started with the Alcator experiment, that use cryogenically cooled normal conductors. The relevant structurals problems have been solved by designing a machine that operates within the material elasticity limits under all foreseen conditions.

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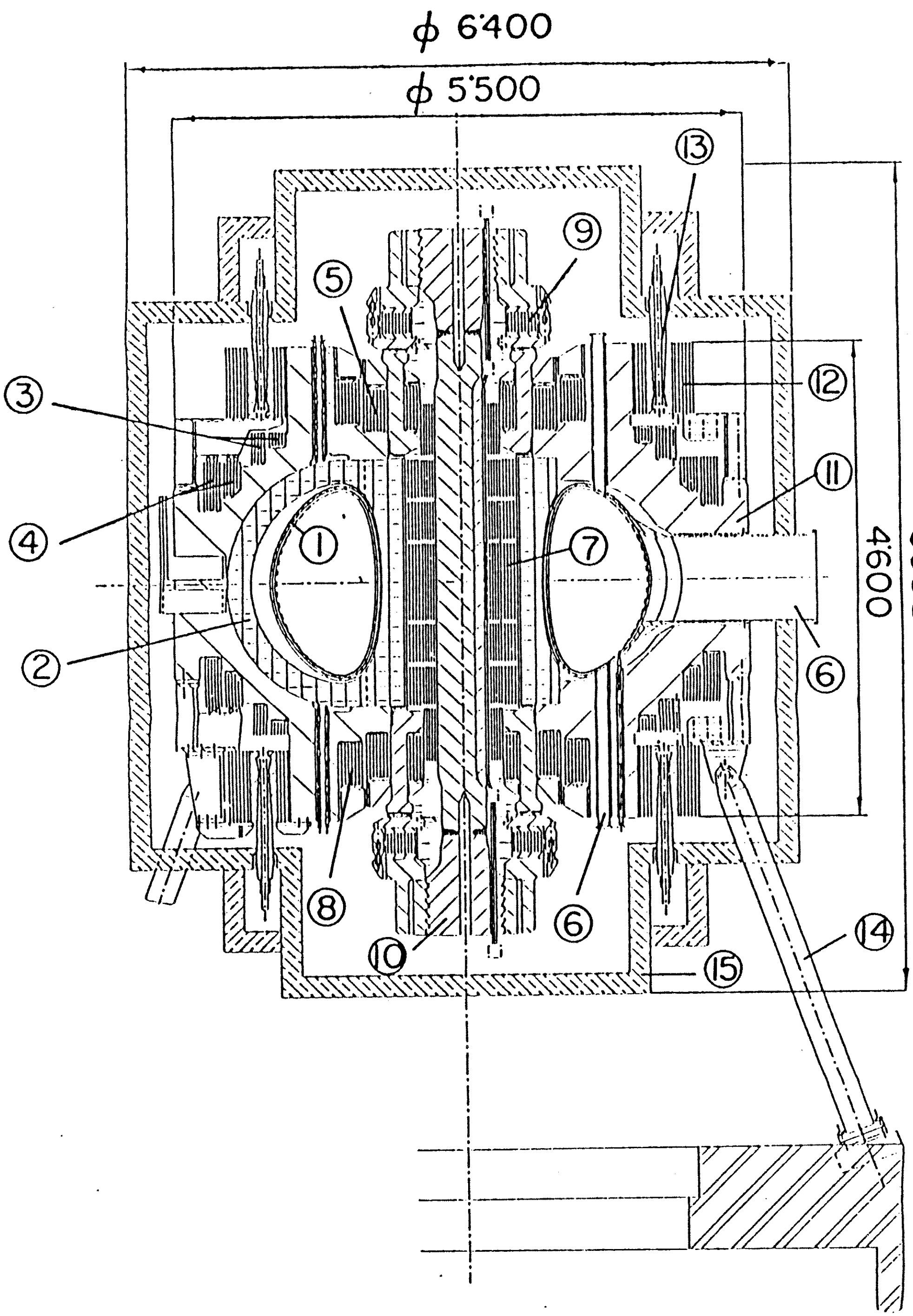

(1) PLASMA CHAMBER
(2) TOROIDAL MAGNET
(3) SHAPING COILS
(4) EQUILIBRIUM COILS
(5) OUTER TRANSFORMER COIL
(6) EQUATORIAL AND VERTICAL PORTS
(7) CENTRAL SOLENOID
(8) SHAPING + TRANSFORMER COIL
(9) AXIAL PRESS
(10) CENTRAL POST
(11) C-CLAMP
(12) SHRINK RING
(13) TENSIONING WEDGES
(14) SUPPORTING LEGS
(15) CRYOSTAT

R_o = 1320 mm
a = 470 mm
b = 870 mm