Highlights of the Columbus Concept

B. Coppi and M.F. Salvetti
Massachusetts Institute of Technology, Cambridge, MA 02139, U.S.

Massachusetts Institute of Technology
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1. Introduction

An important problem in the physics of high energy plasmas is to understand the stability and transport properties of a self organized thermal D-T fusing plasma under conditions where the heating by the fusion produced particles, with multi-MeV energies, can compensate all forms of energy loss. The condition where the rate of energy transferred from the fusion-produced particles to the thermal plasma is sufficient to compensate for all the forms of energy loss from the plasma is defined to be the real ignition condition. Consequently, when ignition is reached, the so-called thermonuclear instability can develop and this can be regarded as the demonstration of a self sustained fusion reactor. The high magnetic field technologies adopted for an experiment like Ignitor, the first proposed and designed to reach ignition and the only magnetic confinement concept maintaining this goal at this time, and the physics that it is expected to uncover, are directly relevant to the design of future fusion reactors [1].

The Columbus device is proposed as one component of a spectrum of experiments that are needed to explore the physics of fusion burning plasmas up to ignition. Columbus, which has a larger volume than Ignitor by about 50%, preserves the ability to confine, under macroscopically stable conditions, plasmas with peak pressures exceeding 3 MPa, corresponding to ignition at central plasma densities around 10^{21} nuclei/m^3 and to reach this regime by ohmic heating alone. In particular, the Columbus program is proposed as a U.S. counterpart to the Ignitor program conducted in Italy and to be complementary to it. The machine costs and its development can be minimized by incorporating the main engineering solutions devised for Ignitor and taking advantage of the results of the R&D effort that has been carried out already.

As will be shown in the following sections the main difference between Columbus and Ignitor is in the magnet and plasma current densities. Thus, longer current pulse times can be produced for the same excursion of the magnet temperature. On the other hand we note that the ratio of the pulse time to the collisional (classical) current redistribution time is improved relative to Ignitor. In fact, the value of this ratio for Ignitor is about the same as that estimated for the ITER-FEAT concept.

The reference design parameters of the machine are given in Table I and compared with those of Ignitor. A comparison of the ratios of the flattop pulse duration to the current redistribution time in the burning plasma machine concepts that have been under consideration recently is given in Table II.

Table I. Reference Parameters of Columbus compared to Ignitor

PARAMETER	Columbus	Ignitor
Major radius R_0	1.50 m	1.32 m
Minor radii $a \times b$	0.535 m × 0.98 m	0.47 m × 0.86 m
Aspect ratio A	2.8	2.8
Elongation κ	1.83	1.83
Triangularity δ	0.4	0.4
Vacuum Toroidal Field B_T at $R = R_0$	$\lesssim$ 12.6 T	$\lesssim$ 13T
Toroidal Current I_p	$\lesssim$ 12.2 MA	$\lesssim$ 11 MA
Poloidal Current I_θ	$\lesssim$ 10 MA	$\lesssim$ 9 MA
Paramagnetic Field Produced by I_θ	$\simeq$ 1.4 T	$\simeq$ 1.4 T
Mean Poloidal Field $\bar{\bar{B}}_p \equiv I_p / \left(5\sqrt{ab}\right)$	$\simeq$ 3.4 T	$\simeq$ 3.4 T
Confinement Strength $S_c \equiv \bar{\bar{B}}_p I_p$	$\lesssim$ 41.5 MN/m	$\lesssim$ 38 MN/m
Toroidal Current Density $\langle J_\Phi \rangle \equiv I_p / (\pi ab)$	$\lesssim$ 7.4 MA/m^2	$\lesssim$ 9.3 MA/m^2
Maximum Poloidal Field B_{pM} $(R < R_0)$	$\lesssim$ 6.5 T	$\lesssim$ 6.5 T
Edge Magnetic Safety Factor q_Ψ	3.6 @ $I_p \simeq$ 12.2 MA	3.6 @ $I_p \simeq$ 11 MA
Magnetic Flux Swing	$\lesssim$ 37.5 Vs	$\lesssim$ 33 Vs
Plasma Volume V_0	$\simeq$ 14.5 m^3	$\simeq$ 10 m^3
Plasma Surface S_0	$\simeq$ 44 m^2	$\simeq$ 34 m^2

2. Scale Up Criteria

In principle it is possible to scale up the size of Ignitor while maintaining the ability to reach ignition and reasonable margins against the onset of macroscopic instabilities in plasmas with high central pressures. However, in practice all the relevant conditions limit sharply the maximum size of a high field machine of this type that can be realistically constructed.

An additional important asset to be preserved is the strength of ohmic heating, to the extent that ignition can be attained even in the case of a poor performance of the auxiliary heating system, by ohmic heating alone. In fact, the only system demonstrated to be capable of heating the high density plasmas that Ignitor and Columbus can be expected to produce is Ion Cyclotron Resonance Heating, but its reliability has been problematic. Taking all factors into account, the most important parameter guiding the machine design is the value of the mean poloidal field that can be achieved. In particular we consider the average

$$\overline{\overline{B}}_p = \frac{I_p}{5\sqrt{ab}}\,, \qquad (2.1)$$

where I_p is the toroidal plasma current in MA, and a and b are the minor radii of the plasma cross section. We take $\overline{\overline{B}}_p \simeq 3.4$ T as the target value for Columbus, a value that is close to that chosen for Ignitor.

In selecting the parameters for Columbus relative to Ignitor we are guided by the following criteria:

i) to decrease the current densities in both the toroidal and poloidal magnet systems. Reduced current densities allows for longer plasma pulses since the rate of ohmic heating of the magnets is decreased

ii) to have a self-similar geometric configuration, scaled up by the factor 25/22 relative to Ignitor. Therefore, the major radius becomes R_0 = 1.50 m

iii) to maintain the current in each of the plates that form the toroidal magnet coils at the same value as in Ignitor, that is 357.3 kA. The toroidal magnet is made of 24 coils, as in Ignitor, in order to keep the value of the toroidal field ripple within acceptable limits. Therefore, each coil contains 11 plates rather than 10 as in Ignitor.

Then, the total magnet current is $I_M \simeq 94.38$ MA-turn, corresponding to a magnetic field $B_T \simeq 12.6$ T at R_0 = 1.50 m. Consequently the average current density in the coils is reduced by a factor

$$\frac{22}{25} \cdot \frac{12.6}{13} \simeq 85\% ,$$

13 T being the toroidal magnetic field on axis (R = 1.32 m) for Ignitor. We note that the length of the pulse over which the current in the coils produces the same temperature rise as a result of Ohmic heating (Figure 2.1) scales roughly as $1/j_c^2$, j_c being the average current density in the toroidal coils.

An illustrative evaluation of the current rise in the toroidal coils has been carried out by Dr. G. Cenacchi using the FORTE code [2] and the main results are given in Table III. This has led us to identify a reference current pulse in the toroidal magnet (shown in Figure 2.2), assuming that the rise and the decay phases of the current are programmed as in the case of

Ignitor. Thus, the pulse flattop is extended to about 7.6 s. The allowed excursion of the maximum local value of the temperature in the inner leg of the toroidal magnet is taken to be that considered for Ignitor, from 30 K to $\simeq 240$ K. The typical dependence of the ratio of the electrical resistivity to the specific heat for ETP copper that had been chosen originally for the TF coils of Ignitor is illustrated by Figure 2.3, for different values of the magnetic field. The simple field distribution within the toroidal magnet of Columbus is given in Figure 2.4. As in the case of Ignitor, the adoption of OFHC copper for the toroidal magnet plates of Columbus is envisioned. This kind of copper has improved thermo-mechanical properties and leads to a better temperature distribution within the magnets than the ETP copper. Because of the exposure of the Columbus magnets to the high energy neutrons resulting from fusion reactions, a slight decrease of the flattop length should be expected relative to the value given in Figure 2.2.

By adopting

$$a \simeq 0.535\,m \text{ and } b \simeq 0.980\,m$$

as the values of the minor radii, then

$$\overline{\overline{B}}_p \simeq 3.4\,T$$

corresponds to a toroidal plasma current

$$I_p \simeq 12.2\,MA,$$

and a magnetic safety factor q_a about equal to that assumed for Ignitor, i.e. $q_a \simeq 3.6$. It is evident that the plasma current density for Columbus is smaller than that for Ignitor (Table I). Since the particle density limit is proportional to the current density, this will be lower for Columbus. On the other hand, the peak plasma density that is needed to reach ignition remains well below the density limit and the decrease of this is not expected to undermine the performances of Columbus. Clearly, this is not the case for the ITER-FEAT concept whose characteristic low current densities involve also low density limits and the necessity to operate in their proximity. We note that if a safety factor $q_{95}(\psi) \simeq 3.6$ is adopted for the ITER-FEAT concept, with its maximum toroidal field on axis (R_0 = 6.2 m) of 5.3 T, the plasma current that it would produce is 12.5 MA, about the same as that of Columbus. Table II gives additional plasma reference parameters for Columbus.

A relatively simple estimate of the plasma and design parameters of Columbus relative to those of Ignitor can be made on the basis of the ratios χ = 25/22 that represents the change in linear dimension of most machine components relative to Ignitor and ζ = 12.6/13 that accounts for the slight reduction of the toroidal magnetic field in Columbus relative to Ignitor. The increased dimensions of Columbus relative to Ignitor allow a higher degree of access to the plasma for diagnostics, pellet injection, RF antennas, vacuum pumping system, remote maintenance, etc.

Table IV introduces a comparison among relevant parameters of presently proposed fusion burning plasma experiments. Table V shows a comparison with the ITER-Feat concept for the same value of the magnetic safety factor ($q_{95}(\psi) \simeq 3.6$). It is evident that Columbus and Ignitor are the "largest" among the presently proposed experiments in terms of number of orbits of the thermal nuclei inside the minor radius.

Table II. Additional Reference Plasma Parameters for Columbus

PARAMETER	VALUE
Plasma thermal energy W (MJ)	$\simeq 17.4$
Volume average temperature $\langle T \rangle$ (keV)	$\simeq 3.5$
Volume average density $\langle n \rangle$ (m^{-3})	$\simeq 5\times10^{20}$
Poloidal beta β_{pol}	$\simeq 0.2$
Internal conductance per unit length l_i	$\simeq 0.75$
Total plasma inductance $L_p^{tot} = \mu_0 R_0 \left[\ln\left(\frac{16 R_0}{(1+\kappa)a} \right) - 2 + \frac{l_i}{2} \right]$ (μH)	$\simeq 2.14$
Plasma internal magnetic energy $E_M^{(\mathrm{int})} = \frac{1}{2}\left(\mu_0 R_0 \frac{l_i}{2} \right) I_p^2$ (MJ)	$\simeq 52.6$
Plasma total magnetic energy $E_M^{tot} = \frac{1}{2} L_p^{tot} I_p^2$ (MJ)	$\simeq 159$

Table III. Reference operation scenario: Excursion of the maximum local temperature

Time (s)	Ohmic Energy Input [MJ]	Ohmic Power [MW]	Magnetic Energy- TF Cavity [MJ]	T_{max} [K]
4.0	47.7	37.2	428	73.9
5.5	96.6	30.3	431	86.7
8.0	284	130	1250	131.0
9.0	407	122	1250	144.7
15.6	1480	224	1100	198.5
18.2	1760	44.5	0	221.6

Table IV. Relevant parameters of proposed fusion burning plasma experiments

RELEVANT PARAMETERS		ITER	FIRE	Ignitor	Columbus
		@ $q_a = 3$			
Pulse flat top	t_{pulse} (s)	400	20	6	11.4
Criticality parameter	$K_f = P_{alpha} / P_{Losses}$	2/3	2/3	1 [a)]	1
Minor radius	a (m)	2	0.595	0.47	0.535
Peak el. temperature	T_{e0} (keV)	25	13	11.5	11.5
Profile parameter	α_T (parab)	1	1	2	2
Purity parameter	Z_{eff}	1.7	1.4	1.2	1.2
Current redistribution time parameter	$\tau_{cr}^{coll} \propto \dfrac{a^2 T_{e0}^{3/2}}{Z_{eff}} \dfrac{1}{\left(1+(3/2)\alpha_{T,parab}\right)}$ b)	118	4.7	1.8	2.33
Relevant parameter of comparison	$\propto \dfrac{t_{pulse}(s)}{\tau_{cr}^{coll}}$	3.4	4.2	3.3	4.9

a) Ignition: onset of the thermonuclear instability; b) Freidberg Report

Table V. Relative size comparison of Columbus and ITER-Feat, and Ignitor and ITER-Feat

RELEVANT PARAMETERS		VALUE
Plasma Current	I_p	
Poloidal ion gyro radius	$\rho_{bi} = \frac{v_{thi} m_i c}{e\bar{\bar{B}}_p}$	
Number of orbits of thermal nuclei	$\mathcal{L}_p = \frac{\bar{a}}{\rho_{bi}} \propto \frac{I_p}{\sqrt{T_{i0}}}$	
Parameter of comparison between Columbus and ITER-Feat	$\frac{\mathcal{L}_p \mid_{Columbus}}{\mathcal{L}_p \mid_{ITER}}$	$> \frac{12.2}{12.5}\sqrt{2} = 1.38$
Parameter of comparison between Ignitor and ITER-Feat	$\frac{\mathcal{L}_p \mid_{Ignitor}}{\mathcal{L}_p \mid_{ITER}}$	$> \frac{11}{12.5}\sqrt{2} = 1.24$

$\bar{a} = \sqrt{ab} = mean\ minor\ radius$

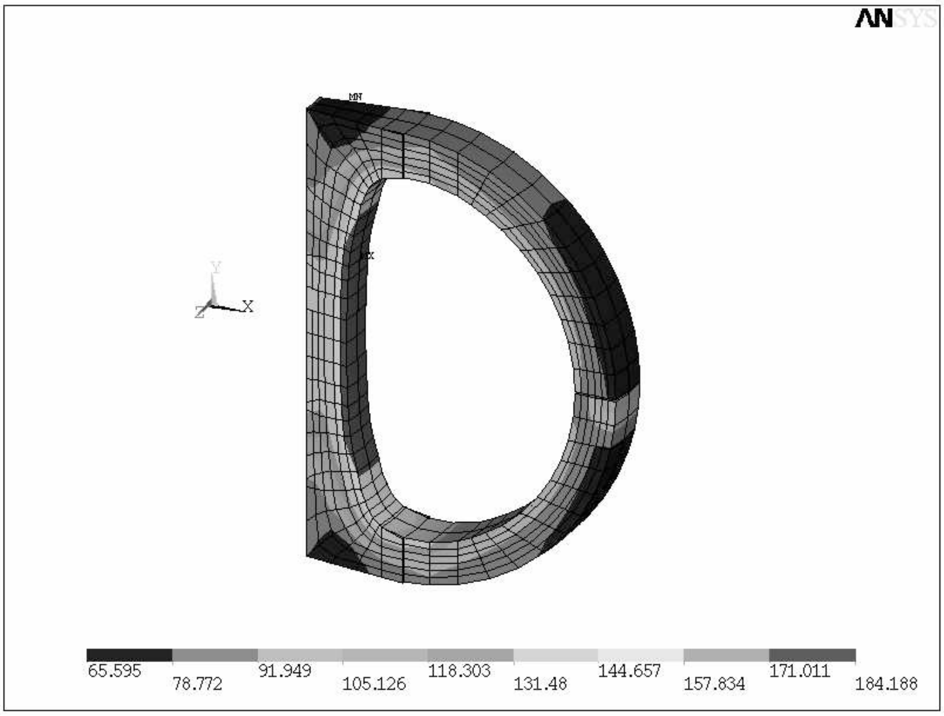


Figure 2.1: Temperature distribution [K] obtained at the end of the plasma pulse assuming a rise of the magnet current that lasts 11.2 s and a flattop that lasts 5.6 s (neglecting neutron energy deposition).

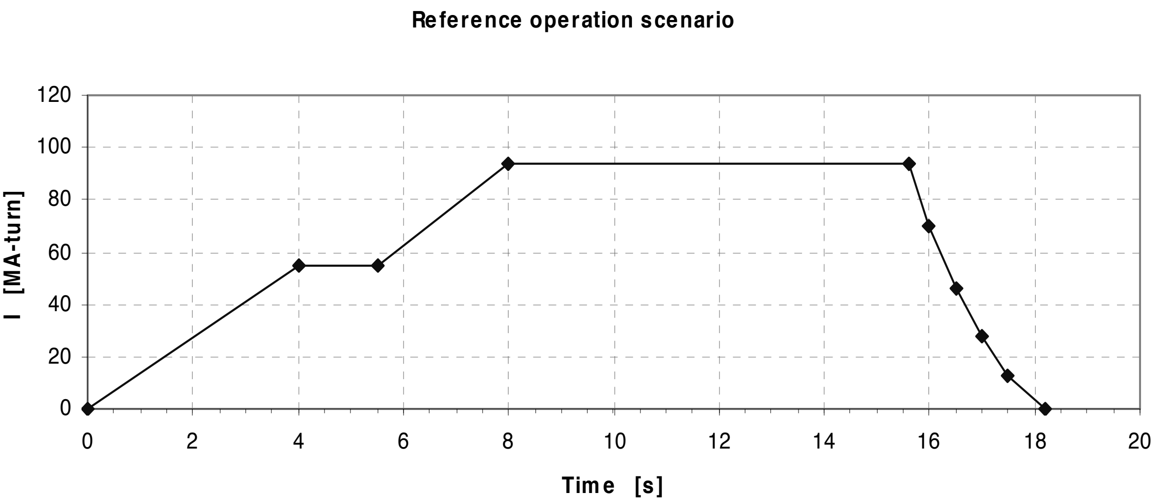


Figure 2.2: Evolution of the total current of the toroidal magnet (MA-turn) for the reference operation scenario. Courtesy of G.Cenacchi (2003).

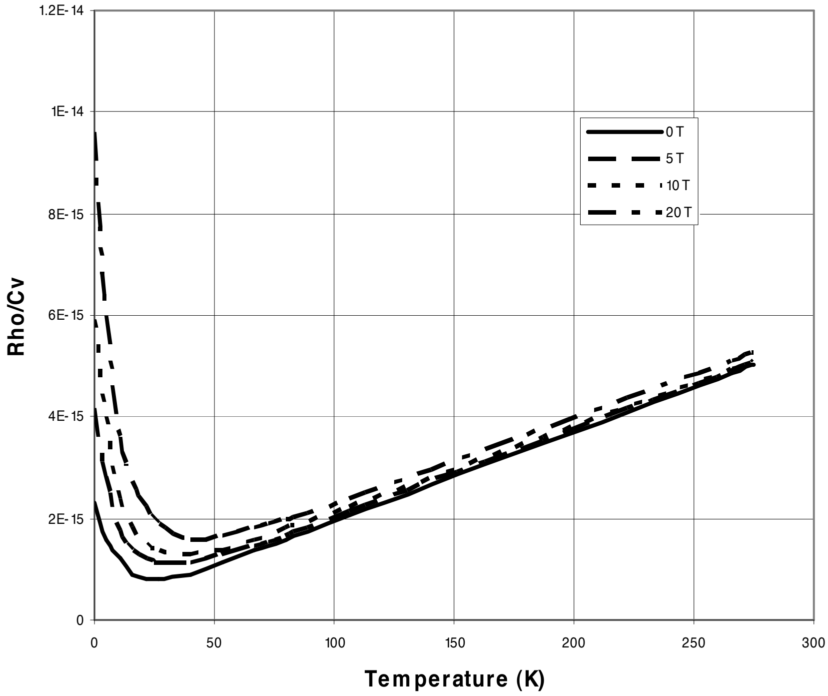


Figure 2.3: Ratio of electrical resistivity to specific heat of ETP copper as a function of temperature, for different magnetic field values.

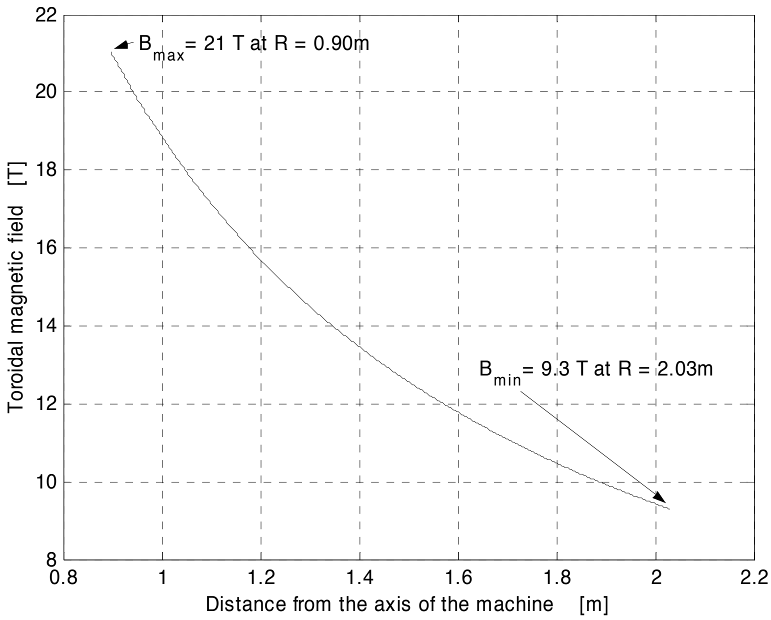


Figure 2.4: Toroidal magnetic field distribution within the cavity of the TF coils of Columbus for the reference design parameters and neglecting the effects of the plasma currents.

$$L_p = \frac{\bar{a}}{\bar{\rho}_{bi}} \propto \frac{I_p}{\sqrt{T_{i0}}} \qquad \bar{a} = \sqrt{ab} = mean\ radius$$

$$I_p = 5\bar{a}\bar{\bar{B}}_p \qquad \rho_{bi} = \frac{v_{thi} m_i c}{e\bar{\bar{B}}_p}$$

$$\frac{L_p\big|_{Ignitor}}{L_p\big|_{ITER}} > \frac{11}{12.75}\sqrt{2} = 1.22 \qquad \frac{L_p\big|_{Columbus}}{L_p\big|_{ITER}} > \frac{12.2}{12.75}\sqrt{2} = 1.35$$

Figure 2.5: Number of orbits of the thermal nuclei in Columbus and Ignitor compared to the ITER-Feat concept when $q_{95}(\psi) \simeq 3.6$

3. Machine layout

Columbus, like the Ignitor device, is characterized by the complete integration of its major components: toroidal field system, supporting mechanical structure, poloidal field system and vacuum vessel. In particular, locating the central solenoid outside the toroidal field magnet cavity is considered more reliable than placing it inside and has been more extensively tested in the high field machines, starting with Alcator A, that have been constructed so far.

Bucked and wedged structure

The design of the Columbus device maintains the structural concept of the bucked and wedged toroidal magnet for Ignitor. This consists of an optimized combination of "wedging" between well defined areas of the TF copper magnets and the magnet reinforcing mechanical structures (C-Clamps) and "bucking" between the TF coils and the split Central Solenoid (CS), as shown in Figures 3.1 and 3.2. In addition, the CS is mechanically coupled to an inner "bucking" post made of steel (Central Post). The bucked and wedged solution represents the most effective way to optimize the structural behavior of the machine and to utilize its mechanical capabilities. This solution requires a careful and precise fit up between the TF and the CS. However, modern laser surveying techniques and the numerically controlled machining equipment now available make the fit up of the coils a feasible operation.

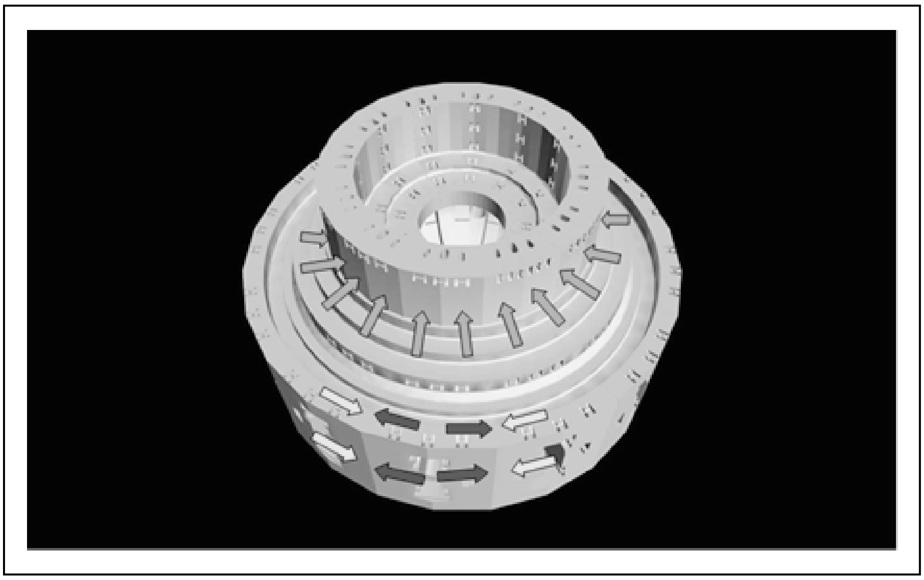


Figure 3.1: C-clamp assembly of the Ignitor machine. The arrows represent the action of the preloading ring on the 24 C-Clamps and the wedging generated on their outer regions. Courtesy of the Ignitor group (2003).

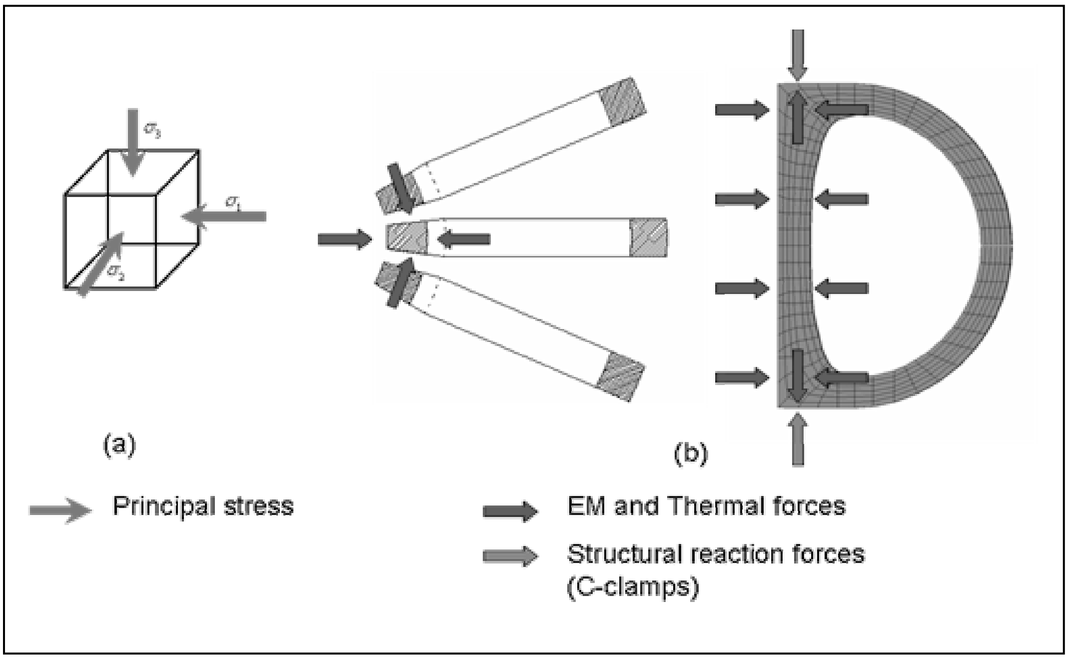


Figure 3.2: (a) Elementary cube on which the principal stresses σ_1, σ_2, and σ_3 are applied, (b) qualitative sketch describing the forces applied on the inner leg of the TF coil of Columbus.

4. Toroidal field system

The TF system is composed of 24 identical coils, as shown in Figure 4.1. The material chosen to fabricate the magnet plates is copper OFHC. The TF coils start from the optimal temperature of 30 K in order to maximize the useful current duration. This corresponds to the lowest ratio of the electrical resistivity to the specific heat for the adopted type of copper (Figure 2.2). As in Ignitor, the coil temperature at the end of the pulse will be limited to about 240 K, in order to maintain an adequate margin against the stresses within the insulating material between the magnet plates, which remains rather cold during the pulse.

The shape of the toroidal magnet cavity is chosen to match the elongated cross section of the plasma column, in order to minimize the out-of-plane forces and twisting moments generated during the plasma operations when the extended limiter configuration of the plasma column is being produced.

The electrical resistance, inductance, and time constants of the toroidal magnets will change during the pulse because of the temperature increase within the conductor, the magneto-resistive effect, and the skin effect that is particularly pronounced in the inner leg of the TF coils.

The TF coil filling factor, i.e. the fraction of the equatorial magnet cross section that is filled by copper, has a minimum of about 0.92 at the CS side of the magnet inner leg and gradually increases to a maximum value (0.95) on the outboard side.

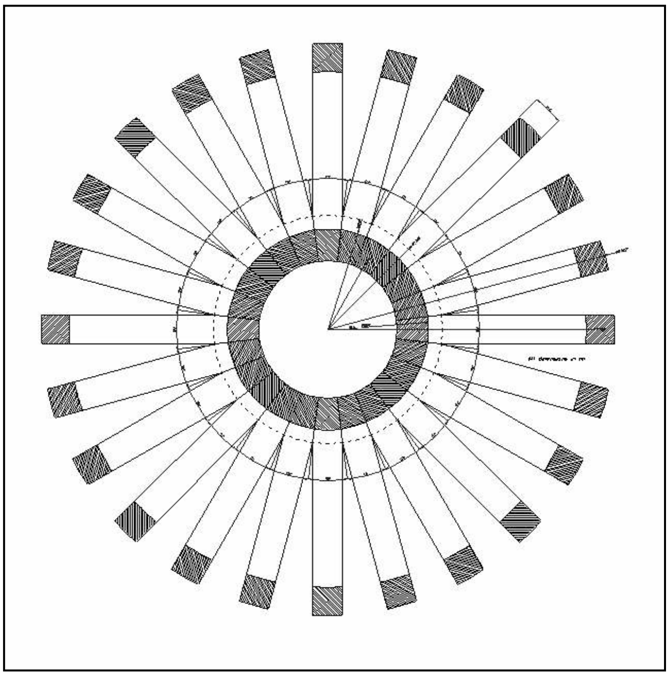


Figure 4.1: Toroidal Field (TF) system of the Columbus machine. The array consists of 24 identical copper coils each of them consisting of a stack of copper plates.

Permanent mechanical press and electromagnetic press systems

The necessary machine structural performance is obtained by designing the copper coils and the structural elements (C-Clamps, central post, bracing rings) in such a way that the system, with the aid of an electromagnetic radial press when necessary, can withstand the forces generated within the magnets. As in Ignitor, the set of 24 stainless steel C-Clamps forms a complete shell which surrounds the 24 TF coils. These coils are pre-stressed through the C-Clamps by means of a permanent mechanical press system (two bracing rings, one on the top and one on the bottom of the machine) that creates a vertical pre-load on the inner legs of the TF coils (Figure 4.2). A representation of the C-clamp array of Ignitor, which is also a qualitative description of the Columbus C-clamp system, is shown in Figure 3.1. The electromagnetic press consists of two pairs of concentric poloidal coils with opposite currents, symmetrically located relative to the machine equatorial plane. The press acts to maintain, as closely as possible, a hydrostatic stress distribution within the TF inner legs, minimizing the von Mises equivalent stresses. This structural solution ensures that the inner legs of the TF coils possess a sufficient degree of mechanical strength to withstand the electrodynamic stresses, while allowing enough deformation to cope with the thermal expansion that occurs during the plasma discharge.

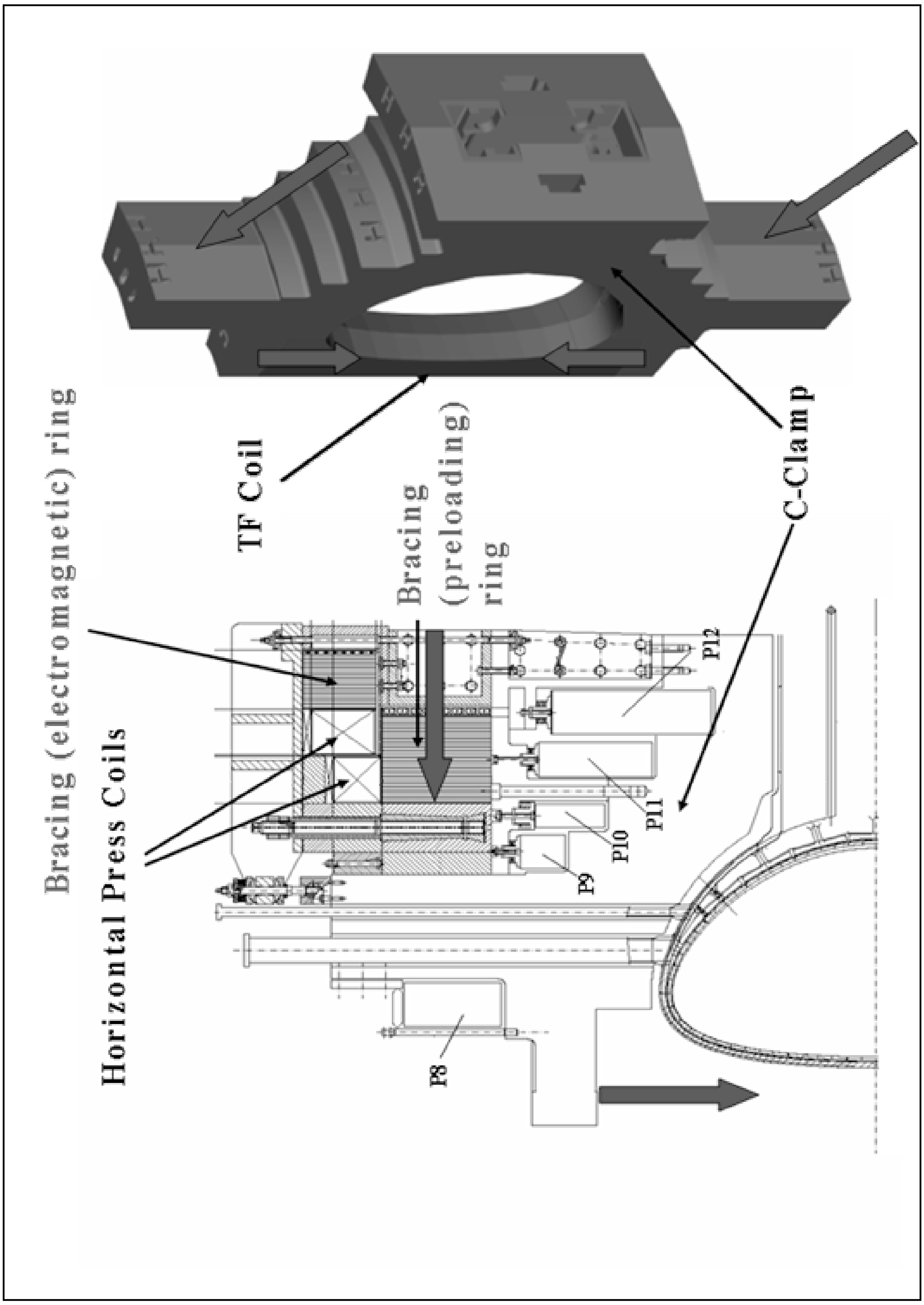


Figure 4.2: Permanent vertical preloading on the TF coils of the Ignitor machine. The same solution is adopted for the Columbus concept. The arrows represent the actions of the Preloading Ring on the C-Clamps and those of the C-Clamps on the TF coil. Courtesy of the Ignitor group (2003).

5. Poloidal field system

The Poloidal Field System of Columbus is designed to

i) generate the magnetic flux variations linked with the plasma column necessary to induce the required plasma currents

ii) produce the desired sequence of plasma equilibrium configurations. When the extended limiter configuration is to be generated this has to fit the profile of the First Wall. When the double null configuration is to be attained the relevant X-points have to be produced at the most appropriate location near the First Wall

iii) ensure the stability of the plasma column against radial and vertical displacements.

The required poloidal fields are produced by an optimized system of 12+12 coils located symmetrically relative to the machine equatorial plane. The main components of the Poloidal Field System are the Central Solenoid and the Outer Coil Assembly.

The primary function of the Central Solenoid is to produce most of the magnetic flux variation required to drive the prescribed plasma current. The optimized design of the system subdivides the Central Solenoid into two concentric groups of coils to maximize the magnetic flux variation while keeping the coils within realistic thermal-mechanical limits. The Lorentz forces acting on the inner coils of the Central Solenoid are minimized by driving the lowest possible current densities within those coils, where the magnetic field is highest. The vertical subdivision of the solenoid into four groups of coils ("split solenoid") stacked along the length of the TF Coils inner legs is motivated by the necessity to control effectively the plasma configuration as the plasma parameters evolve toward ignition. The design solution also provides the Central Solenoid with sufficient flexibility to cope with the gradual, but significant, change of stiffness along the height of the inner legs of the TF Coils, as they heat during a pulse.

The Outer Coil Assembly is made of a set of coils that perform different functions depending on their position. Those coils located between the Central Solenoid and the major radius of the machine R_0, coils P7 and P8 in Figure 5.1, provide the main shaping for the plasma configuration, and induce the plasma elongation, while contributing to the required magnetic flux variation. On the outer side of the C-clamp array, the coils located at the highest axial coordinates away from the mid plane, P9 and P10, control the triangularity and the vertical position of the plasma. The coils located furthest from the axis of symmetry, P11 and P12, ensure the horizontal plasma equilibrium and contribute to the creation of a multipolar field configuration at the initial plasma breakdown.

The Central Solenoid of Ignitor is shown in Figure 5.2. The currents driven in the Central Solenoid and in the Outer Poloidal Field coils of Ignitor, including the electromagnetic press coils, are shown in Figure 5.3 for comparison.

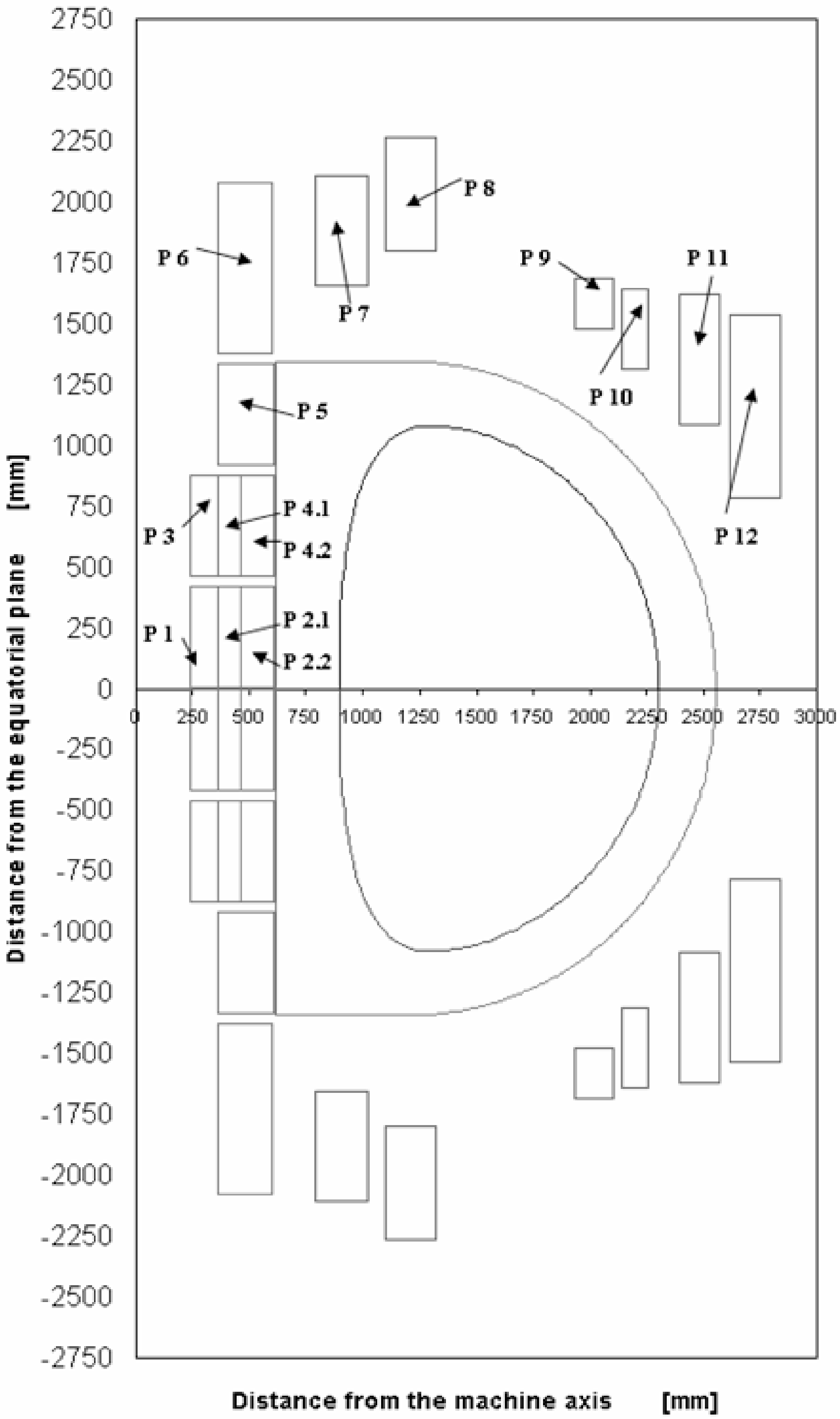


Figure 5.1: Central Solenoid Coils (P1-P6) and Outer Poloidal Field Coils (P7-P12).

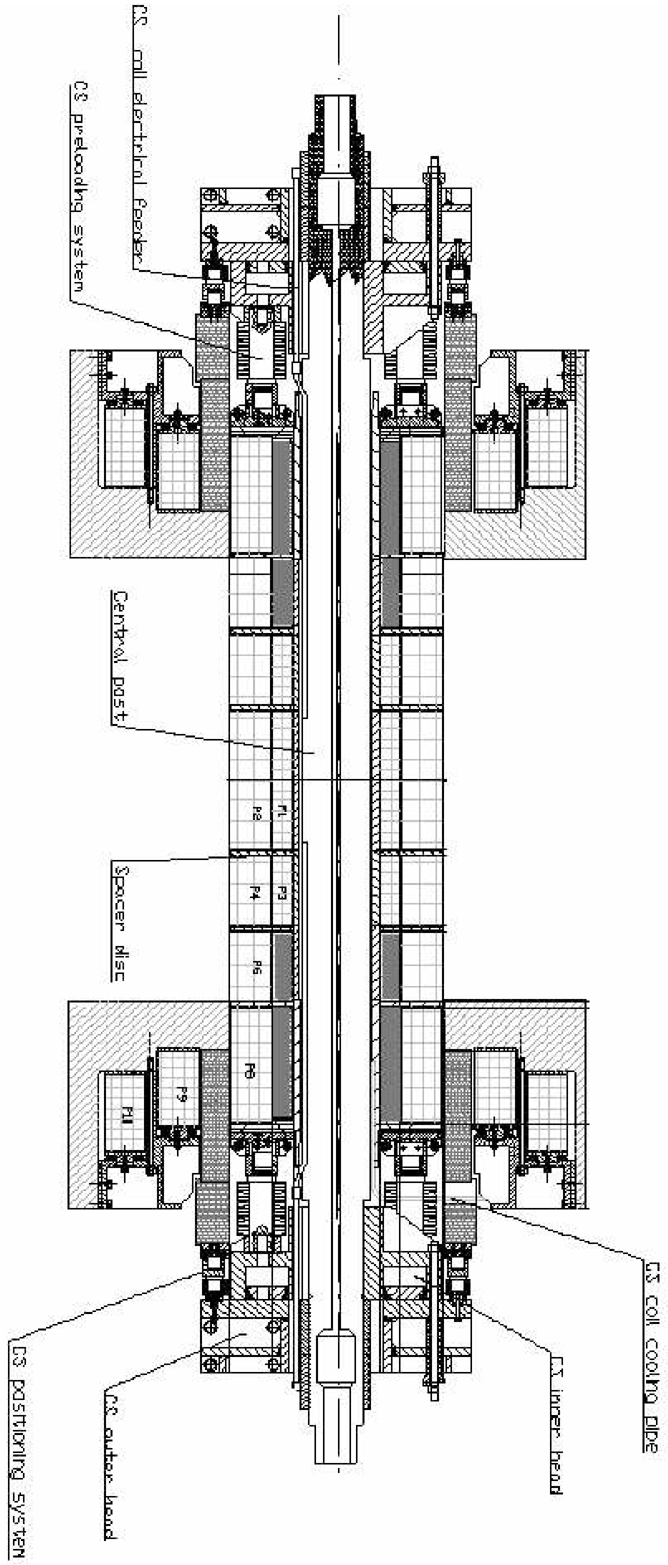
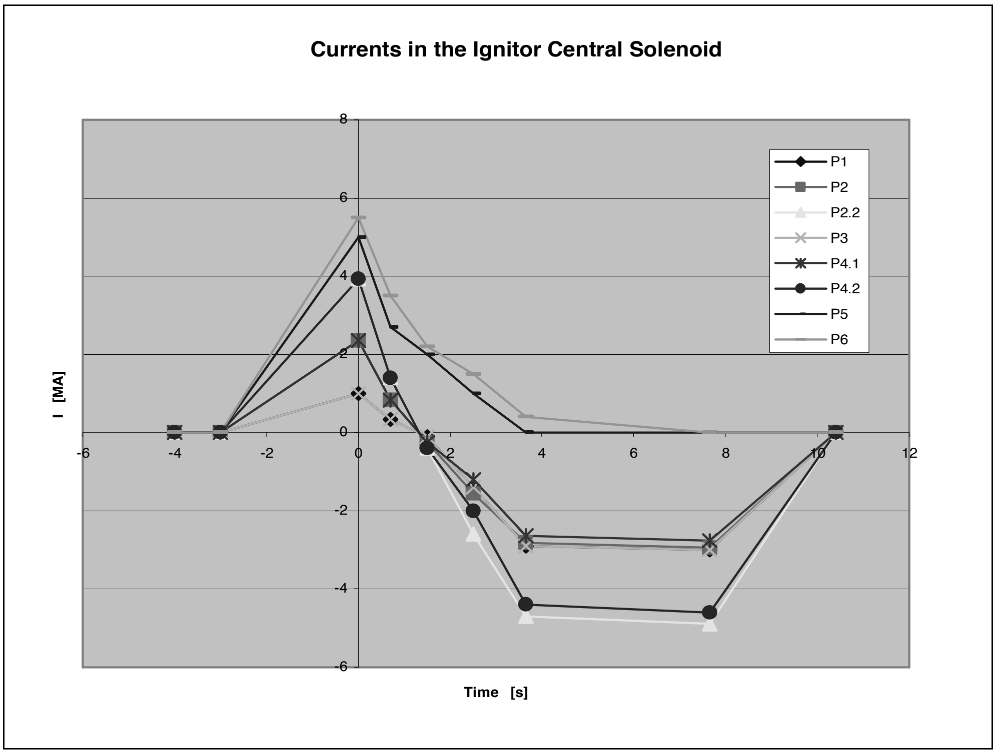


Figure 5.2: Central solenoid of the Ignitor machine. Courtesy of the Ignitor group (2003).



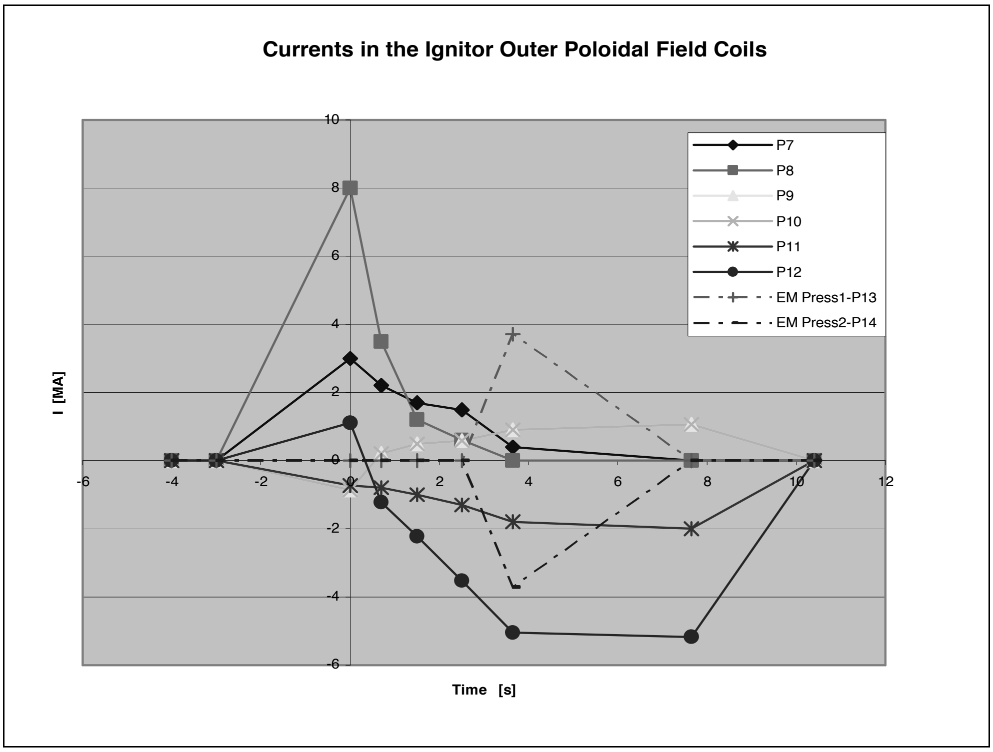


Figure 5.3: Currents in the Ignitor Central Solenoid and in the Outer Poloidal Field Coils. Courtesy of G.Ramogida and M.Roccella (2003).

6. Plasma chamber and First Wall systems

The design of the plasma chamber of Columbus is subjected to a number of requirements related to the vacuum to be produced in it and to the static and dynamic structural loads that can be applied to it. Thermal stresses within the chamber structure are greatly reduced by adopting, like in the case of Ignitor, hydro-mechanical bumper interfaces with the C-Clamps. They consist of radial supports whose stiffness can be controlled by means of hydraulic clamping sleeves. During the plasma pulses the bumpers are locked to increase the rigidity of the plasma chamber. This condition allows the plasma chamber to withstand the dynamic loads generated during disruption events by sharing the asymmetric radial loads between adjacent ports. At the end of each plasma pulse the connection is unlocked to allow for the thermal expansion of the plasma chamber. The system designed for Ignitor is shown in Figure 6.1. Lateral supports are also provided between the equatorial ports and the C-Clamps taking into account the out-of-plane forces generated during disruption events and the thermal expansions of the ports.

The largest loads are expected to be produced by plasma Vertical Displacement Events (VDE) and by plasma disruptions. The expected stresses during normal plasma conditions are considerably lower. These are generated by the external atmospheric pressure ($\simeq 0.1$ MPa), the electromagnetic pressure originated by external field variations, and the secondary stresses induced by the thermal loads on the chamber.

The plasma chamber fills the maximum possible volume within the TF coils, leaving some free space on the outboard side between the First Wall (FW) facing the plasma and the walls of the chamber, to accommodate in-vessel components, such as the ICRH antennas and the pumped limiter system. The plasma chamber has vertical and equatorial access ports for the plasma diagnostics, the vacuum system, the pellet injector, the auxiliary heating system, the in-vessel remote maintenance system, etc. The entire structure of the plasma chamber is envisioned to be fabricated with Inconel 625, a nickel-chromium alloy that ensures very high mechanical strength even at low temperatures and high electrical resistivity that helps to reduce the flux consumption during the plasma start up.

The First Wall consists of molybdenum tiles lining the plasma chamber and acts as an extended toroidal limiter. It covers the entire surface of the plasma chamber, with the exception of the port openings on the outboard side. The molybdenum tiles are supported by multiple Inconel 625 back plates that are connected to the plasma chamber by means of appropriate bolts and can be removed and replaced by the remote handling system. No active cooling for the First Wall is provided; passive cooling takes place by conduction through the back plates to the plasma chamber and by radiation.

Experiments have shown that attaining high density plasmas is more important for good impurity screening than having a classical divertor system included in the machine design [3]. High density plasmas have higher neutral particle density and lower temperature at the plasma edge. In these regimes, in the absence of plasma transport barriers, the level of impurity contamination has consistently been found to be low by a variety of experiments over the last 25 years. Furthermore, recent experiments [4] indicate that, at higher densities, particle recycling from the main chamber and cross field diffusion directly to the surrounding walls play an increasingly dominant role, while the divertor is no longer the main power and particle sink. Thus, as in Ignitor, a classic divertor with coils located inside the toroidal magnet cavity has not been included in the Columbus design. In particular, the easier accessibility to the H-mode ensured by a classical divertor does not compensate for the added complexity and degradation of global plasma parameters associated with the reduction of useful plasma volume. The possibility of spreading the heat loads over a larger surface area by keeping a well-shaped plasma close to the First Wall is still considered the optimal solution for high density plasma regimes.

Nevertheless, the flexibility of the adopted Poloidal Field System can be exploited to generate magnetic configurations with two X-points ("magnetic divertor") and this option, for the plasma equilibrium, is considered in parallel to that of the extender limiter configuration (no X-points within the plasma chamber). With the X-points configurations the appearance of more localized thermal loads must be taken into account. In particular, more complex solutions for the First Wall components can be envisaged, given that the space available for this is slightly increased relative to the case of Ignitor.

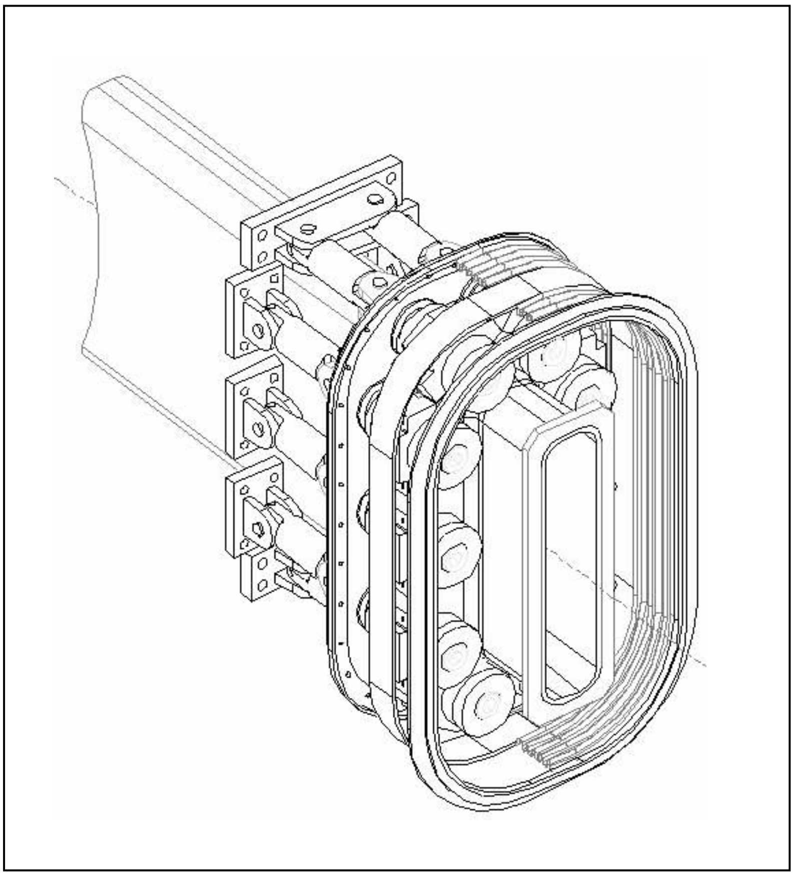


Figure 6.1: Set of ten hydro-mechanical bumper interfaces that connect the equatorial ports of Ignitor to the C-Clamps. Courtesy of the Ignitor group (2003).

7. External Systems

Auxiliary heating system (ICRH)

A system for the injection of radio frequency power at the ion cyclotron frequency (≈100-140 MHz) is included in the machine design in order to gain significant control over the evolution of the temperature and the current density profiles and to shorten the time needed to reach ignition. In non-ignited plasmas, it is also possible to operate at higher temperatures and lower densities in order to increase substantially the α-particle pressure gradient and enhance the severity of α-particle driven modes in order to make their analysis easier.

Cooling system

The useful plasma pulse duration is maximized by cryogenically cooling the copper windings of the magnet coils. As stated earlier, all components, with the exception of the plasma chamber, are cooled before each plasma pulse to an optimal temperature of 30 K by means of He-gas. Given the large number of superconducting magnets operated in recent years with liquid and super-critical Helium, the reliability of the refrigeration plants based on He-gas is well proven. The cooling process is facilitated by the complete integration of all the main components of the device and by the resulting relatively high degree of thermal conductivity that this involves. The entire machine is enclosed in a vacuum insulated cryostat and appropriate vacuum-tight and electrically insulated feedthroughs are provided for the inlet and outlet of Helium and for the electrical connections.

Pellet injector

A pellet injector, with velocities up to ≈ 4 km/s that should be sufficient to reach the central region of the plasma column, is considered to be an integral part of the machine design, in order to produce the peaked density profiles that are optimal for fusion burning conditions and to minimize the anomalous ion transport produced by Ion Temperature Gradient driven modes [5]. An injector of D or D-T pellets ($\simeq 4$ mm diameter) is included in addition to the well-tested technique of gas injection. Moreover, the pellet injector can be used to promote the formation of internal transport barriers [6], for time dependent burn control, and for diagnostic purposes.

8. Structural requirements: loads on the TF coils

The TF coils are the structural elements of Columbus that are subjected to the most complex stress distributions. Clearly, both the electromagnetic and the thermal loads must be adequately withstood to ensure the mechanical reliability of the device and the capability to operate with sufficiently long pulses. The interaction between the currents flowing in the TF coils and the toroidal magnetic field that they produce generates vertical separating forces, inward radial loads on the inner legs of the TF magnets and outward radial loads on the outer legs of the TF magnets. These loads are described as in-plane loads and their distribution is symmetric relative to the equatorial plane. In addition, twisting out-of-plane loads can be generated by the interaction of the TF coil currents and the poloidal magnetic fields generated by the plasma current induced by the Central Solenoid and the Outer Coil Assembly.

The capability of structures to withstand the applied loads is measured usually in terms of an equivalent stress derived from a material failure theory. The von Mises equivalent stress assumes a strain energy failure criterion for the material and is the basis for most elastic-plastic stress considerations and non-linear material models. The Tresca equivalent stress relates to failure due to a maximum shear stress. The Tresca criterion is simpler, more conservative, and is used by the ASME code and some fusion design criteria. The von Mises criterion is used in fusion research, particularly when considering non-linear material properties. Both criteria rely on computing a unique local stress value that describes the real three-dimensional local stress tensor. The equivalent stress becomes the term of comparison to the conventional uniaxial tensile properties of the material that are measured on specimens in the laboratory and are descriptive of the material's mechanical behavior.

The conditions at which the plastic yielding of a material begins or the ultimate tension stress failure (UTS) occurs relate to the difference in the directional components of the principal stresses. The von Mises equivalent stress is defined as

$$\sigma_{eq} = \frac{1}{\sqrt{2}} \sqrt{(\sigma_1 - \sigma_2)^2 + (\sigma_1 - \sigma_3)^2 + (\sigma_2 - \sigma_3)^2} \qquad (8.1)$$

where σ_1, σ_2, σ_3 are the principal stresses[1] (Figure 3.2a).

The most effective way to improve the mechanical behavior of the TF magnets, as inferred from Eq. (8.1), consists of balancing the primary stresses in the radial, toroidal and vertical directions. The goal is to approach an ideal hydrostatic stress state in which a uniform compression is experienced in any of the three directions [7].

This concept is the cornerstone of the structural design of Columbus, as it is for Ignitor. An enlightening example is the behavior of the inner leg of the TF coil (Figure 3.2b). During the plasma pulse the inner leg of the TF coil experiences compressive stresses in two directions: compression in the toroidal direction, caused by the wedging action between the TF coils, and compression in the radial direction, caused by the bucking between the Central Solenoid and the TF inner leg. The third stress component is tensile and is directed along the vertical axis of the machine. The local von Mises equivalent stress can only be reduced by providing a support

[1] Any state of stress can be represented by defining a principal coordinate system the axes of which are perpendicular to the planes on which (a) the maximum normal stresses are applied and (b) the shearing stresses are null. The stresses normal to the principal planes are defined as *principal stresses*.

against the electromagnetic separating forces and the thermal expansion forces along the vertical direction, turning them into compressive forces or at least minimizing them. In Ignitor and Columbus, this task is accomplished by the bracing rings and by the active horizontal press, which is added when the imbalance in the stress components becomes excessive (Figure 8.1). The wedge pressure created by the TF inner leg centering forces supports the twisting out-of-plane forces acting on the coils. The wedge pressure frictionally couples the surfaces of the inner TF legs, the behavior of which becomes similar to that of a large heavy walled torque cylinder [7]. The use of large preloading compression rings and mechanical jacks guarantees a sufficient value of the wedge pressure during all the operating conditions.

A permanent vertical preload on the TF coils is applied by means of two bracing rings. Figure 4.2 shows the process through which the preloading is generated on the TF coils in the Ignitor machine. The identical concept is adopted in Columbus. During the assembly phase, the average temperature of the two preloading rings is increased temporarily in order to expand them and facilitate their positioning around the C-Clamp array. Once the thermal equilibrium between the structures is reached, the rings provide a nearly uniform radial pressure on the C-Clamps. A set of mechanical jacks allows the regulation of the applied pressure. Under the radial load, the C-Clamps act as hinges on the TF coils along the vertical direction and generate a vertical preload on them. Concurrently, the action of the preloading rings creates the wedging between the outer regions of the C-Clamps, as shown in Figure 3.1. It is important to note that the wedged area of the C-Clamps is limited to their external regions. If this were not the case, then the C-Clamps, constrained by the increased toroidal pressure, would not act as flexible joints and the radial preload would not be transformed into the vertical preload on the TF coils.

A supplemental vertical preload is also applied by the action of the radial electromagnetic press when needed. This differs from that of the preloading rings, as the preload generated on the TF coils is not permanent. The combination of the preloading rings and of the electromagnetic press can make the structural behavior of the TF system of Columbus very flexible. In fact, the vertical preload on the TF coil can be modulated according to the desired operating scenarios and the expected thermal expansion of the magnets.

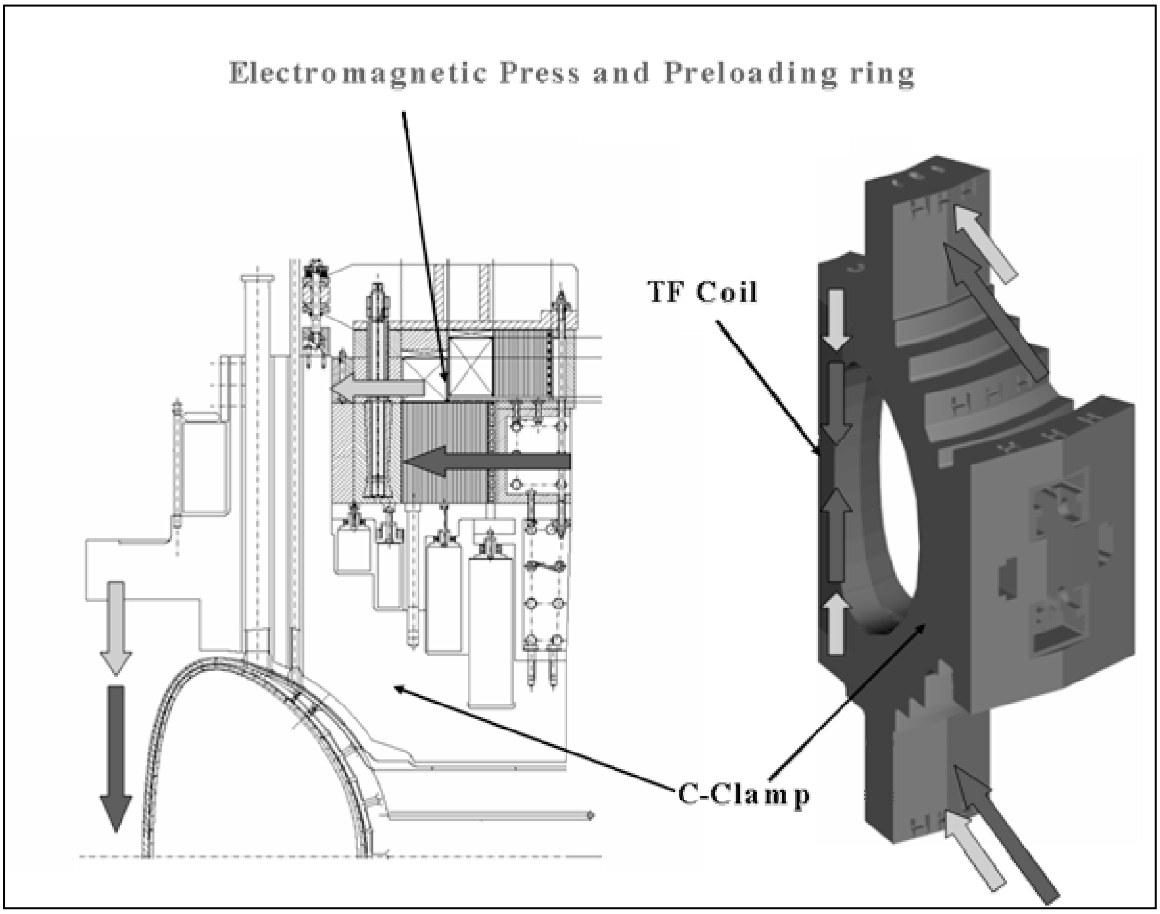


Figure 8.1: Combined action of the Electromagnetic Press and the Preloading Ring on the Ignitor machine. Courtesy of the Ignitor group.

9. Conclusions

The features of the Columbus device have been briefly described. The considerations underlying its choice of parameters are given and compared to those for the Ignitor device. Columbus is an Ignitor-like machine characterized by modestly increased linear dimensions and by lower current densities in both the TF and PF magnets and within the plasma column. It is proposed as a U.S. counterpart to the ongoing Ignitor program. Like Ignitor, the machine is based on normal conducting, cryogenic magnet technologies and an optimized plasma confinement configuration which allow it to reach (real) ignition conditions by fusion reactions. The device concept takes advantage of the maturity of the Ignitor design for the definition of its main components. The linear increase in dimensions by a factor of 25/22 relative to Ignitor corresponds to an increase of the plasma column volume of about 45% and is guided by the criterion that the average poloidal filed which can be produced by the plasma current remains about equal to that of Ignitor for comparable values of the magnetic safety factor. The toroidal magnetic field is slightly decreased, by the factor 12.6/13 relative to Ignitor.

Acknowledgments

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Quoted References

[1] *Fusion policy Advisory Committee final Report* (U.S. Department of Energy, Washington, DC, September, 1991).

[2] Bittoni, B., Cenacchi, G., Guerri, L., Taroni, A., "FORTE: a Finite Element Code for Temperatures, Magnetic Fields and Forces in a Toroidal Magnet", ENEA Report RT/FIMA(79)1 (1979).

[3] Pitcher, C.S. and Stangeby, P.C., *Plasma Phys. and Control. Fusion* **39**, 779 (1997).

[4] LaBombard, B. et al., *Nucl. Fusion* **40**, 2041 (2000).

[5] Coppi, B., Rosenbluth, M.N., Sagdeev, R.Z., *Phys. Fluids* **10** (1967)582.

[6] Baylor, L.R., Jernigan, T.C., et al., *Physics of Plasmas* **7**, 1878 (2000).

[7] Titus, P.H., *Structural Design of High Field Tokamaks,* M.I.T. Plasma Science Fusion Center Report, PSFC/JA-03-9, April 2003.

Appendix A: Description of the Ignitor experiment [Reprint of M.I.T. (RLE) PTP 02/5 (2002)]

The Ignitor Experiment

Ignitor [1] is the first experiment that has been proposed and designed to achieve physical regimes where fusion ignition occurs under controlled conditions. At the present time, it is still the only one capable of attaining ignition in a magnetically confined plasma. Ignitor is designed so that the burning phase exceeds all the intrinsic physical time scales and it specifically addresses the main issues that should be resolved in present day research on nuclear fusion - demonstration of ignition, the study of the physics of the ignition process, and the heating and control procedures for a burning plasma.

The machine is characterized by an optimal combination of high magnetic field ($B_T \leq 13$T), compact dimensions ($R_0 \cong 1.32$ m), relatively low aspect ratio ($R_0/a \cong 2.8$) and considerable plasma cross section elongation and triangularity ($\kappa \cong 1.83$, $\delta \cong 0.4$). The reference central density of the fusing nuclei for which ignition can be achieved is about 10^{21} m^{-3}. The corresponding line-averaged density is well below the known density limit, which is related to the average plasma current density. The considered plasma current is $I_p \cong 11$MA. Ignition corresponds to ratios of the plasma energy density to the magnetic field energy density that are favorable for macroscopic plasma stability.–Ignition can be achieved by ohmic heating alone shortly after the end of the current rise to 11MA. The peak temperature at ignition is expected to be about $T_{e0} \cong T_{i0} \cong 11$ keV for an energy confinement time $\tau_E \cong 0.6$ sec (see Table I). The First Wall

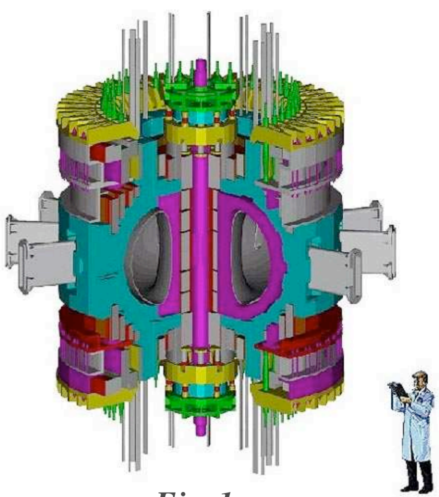


Fig. 1

facing the plasma, made of molybdenum tiles lining the entire plasma chamber, acts as an extended toroidal limiter. The expected peak thermal power loads on the First Wall do not exceed 1.8 MW/m^2 [1] in the standard extended limiter configuration. The poloidal field system of Ignitor can also produce magnetic divertor configurations with two up-down symmetric X-points, at 9 MA and $q_{95}>3$, to facilitate access to the so-called H-mode regime. A preliminary analysis of the thermal loads at the strike points, when the X-points are located near the First Wall, indicates that these loads are acceptable with the First Wall as presently designed.

Experiments have shown that attaining high density plasmas is more important for good impurity screening than including a divertor system in the machine design [2]. High density plasmas have higher neutral particle density and lower temperature at the plasma edge. In these regimes, in the absence of transport barriers, the level of impurity contamination has consistently been found to be low by a variety of experiments over the last 25 years. In fact, the standard view of the divertor as the dominant power and particle sink has been challenged by recent experiments [3], where particle recycling from the main chamber and cross field diffusion in the outer region of the plasma column are observed to play an increasingly important role at higher densities.

A system (ICRF) for the injection of radio frequency power at the ion cyclotron frequency (≅100 – 140 MHz) is included in the machine design. In order to gain significant control over the evolution of the temperature and current density profiles, and to shorten the time needed to reach ignition, less than 5 MW of absorbed power are sufficient, which can be delivered by antennas using 3 of the 12 equatorial ports. In non-ignited plasmas, it is also possible to substantially increase the α-particle pressure gradient and enhance the virulence of α-particle driven modes in order to study them, by operating at higher temperature and lower density and using a higher level of auxiliary heating power, up to 20 MW with 6 antennas. The first exploration of fusion burn conditions in tritium-poor plasmas can also be conducted, with significant production of power from D-^{3}He reactions [4]. Modest ICRF power levels are also adequate, in combination with the ohmic and fusion alpha heating, to access H-mode regimes, according to the available scalings [5]. While these regimes can exhibit longer energy confinement times, they have the disadvantage, for a burning plasma, of featuring flat density profiles. This is one of the reasons why a classic divertor with coils inserted inside the toroidal magnet cavity has not been adopted in the Ignitor design. The easier accessibility to H-modes that this may allow does not compensate for the degradation of global plasma parameters (e.g., the maximum achievable current I_p) and the complexity that such a system, operating in a high magnetic field environment, would involve.

TABLE I: EXAMPLE OF PLASMA PARAMETERS WHEN IGNITION IS REACHED (JETTO CODE)

Toroidal Plasma Current I_p	11 MA
Toroidal Field B_T	13 T
Central Electron Temperature T_{e0}	11.5 keV
Central Ion Temperature T_{i0}	10.5 keV
Central Electron Density n_{e0}	9.5×10^{20} m^{-3}
Central Plasma Pressure p_0	3.3 MPa
Alpha Density Parameter n_α*	1.2×10^{18} m^{-3}
Average Alpha Density $\langle n_\alpha \rangle$	1.1×10^{17} m^{-3}
Fusion Alpha Power P_α	19.2 MW
Plasma Stored Energy W	11.9 MJ
Ohmic Power P_{OH}	11.2 MW
ICRF Power P_{ICRH}	0
Bremsstrahlung Power Loss P_{brem}	3.9 MW
Poloidal Beta $\langle \beta_p \rangle$	0.20
Toroidal Beta $\langle \beta_T \rangle$	1.2 %
Central "safety factor" q_0	$\cong$ 1.1
Edge safety factor $q_\psi = q_\psi(a)$	3.5
Bootstrap Current I_{bs}	0.86 MA
Poloidal Plasma Current	$\cong$ 8.4 MA
Energy Replacement Time τ_E	0.62 sec
Alpha Slowing Down Time $\tau_{\alpha,sd}$	0.05 sec
Average Effective Charge $\langle Z_{eff} \rangle$	1.2

$n_\alpha^* \equiv n_D n_T \langle \sigma v \rangle \tau_{\alpha,sd}$

$\tau_{\alpha,sd} \equiv 0.012 T_{e0}^{3/2}(\mathrm{keV})/n_{e0}\ (10^{20}\ \mathrm{m}^{-3})$

Given the importance that programming the plasma density rise has in order to attain ignition, a pellet injector, with velocities ~ 4 km/s, is an integral part of the machine design. In particular, this injector is to be used to produce the peaked density profiles that are optimal for fusion burning, to minimize anomalous ion transport, to promote the formation of internal transport barriers [6], and for diagnostic purposes.

One of the main criteria for which Ignitor has been designed is to have mean poloidal magnetic fields $\bar{\bar{B}}_p = I_p/(5a\sqrt{\kappa})$ around 3.5 T (see Table II). This is important for macroscopic stability at the high plasma pressures needed for ignition and for allowing the possibility to reach this regime by ohmic heating alone. Recently $\bar{\bar{B}}_p$ has also been identified as the main parameter of merit to assess the performance of a machine magnet system for the confinement of a toroidal plasma [7]. Therefore, given our present knowledge of the macroscopic stability of well-confined plasmas, any larger Ignitor-like device should also attain similar values of $\bar{\bar{B}}_p$.

The machine (Figs.1 and 2) is characterized by a complete structural integration of its major components (toroidal field (TF) system, poloidal field system, central post, C-Clamps and plasma chamber). A "split" central solenoid is adopted to provide the flexibility to produce the expected sequence of plasma equilibrium configurations during the plasma current and pressure rise. The structural concept upon which the machine is based involves an optimized combination of "bucking" between the toroidal field coils and the central solenoid with its central post, and "wedging" between the inner legs of the toroidal field magnet coils and between the C-Clamps in the outboard region. The machine core, consisting of the copper TF coils, the major structural elements (C-Clamps, central post, bracing rings) and the plasma chamber, is designed to withstand the forces produced within it with the aid of a radial electromagnetic press when necessary. The set of stainless steel C-Clamps forms a complete shell, which surrounds the 24 TF coils. These coils are pre-stressed through the C-Clamps by means of a permanent mechanical press system (two bracing rings) that creates a vertical pre-load on the inner legs of the TF coils. This permanent press is supplemented by an electromagnetic press that is activated only at the maximum magnet currents, to maintain as closely as possible a hydrostatic stress distribution in the TF coils in order to minimize the von Mises equivalent stresses. This ensures that the inner legs of the TF coils possess a sufficient degree of mechanical strength to withstand the electrodynamic stresses, while allowing enough deformation to cope with the thermal expansion that occurs during the plasma discharge. The entire machine core is enclosed by a cryostat. All components, with the exception of the vacuum vessel, are cooled before each plasma pulse by means of He gas, to an optimal temperature of 30 K, at which the ratio of the electrical resistivity to the specific heat of copper is minimum.

An important element of the Ignitor experiment is the site where it will operate. The ENEL center of Rondissone, near Turin, has been selected on the basis of its credits. Rondissone is a major node of the European electrical grid and has been analyzed and authorized to accept loads corresponding to the highest plasma currents and fields in Ignitor. Moreover, Rondissone has the unique advantage of housing the large scale test facilities of the C.E.S.I. Center for advanced high current technologies and allowing ready access to the expertise of this Center.

References Appendix A

[1] COPPI, B., , et al., “Critical Physics Issues for Ignition Experiments: Ignitor”, M.I.T. (RLE) Report PTP99/06, Cambridge MA (1999) and references therein.

[2] PITCHER, C.S., and STANGEBY, P.C., *Plasma Phys. and Control. Fusion* **39**, 779 (1997).

[3] LABOMBARD, B., et al., *Nucl. Fusion* **40**, 2041 (2000).

[4] COPPI, B., DETRAGIACHE, P., MIGLIUOLO, S., *Fusion Technology* **25**, 353 (1994).

[5] SNIPES, J.A., et al., *Plasma Phys. Control. Fusion* **42**, 381 (2000).

[6] BAYLOR, L.R., JERNIGAN, T.C., et al., *Physics of Plasmas* **7**, 1878 (2000).

[7] J.H. SCHULTZ at al., “Advanced magnets and implications for BPX”, BPS Workshop II (G.A. San Diego, 2001).

Table II: REFERENCE DESIGN PARAMETERS

Major radius R_0	1.32m
Minor radius $a \times b$	0.47×0.86 m
Aspect ratio A	2.8
Elongation κ	1.83
Triangularity δ	0.4
Toroidal field B_T	$\lesssim$ 13 T
Toroidal current I_p	$\lesssim$ 11 MA
Maximum poloidal field $B_{p,\max}$	$\lesssim$ 6.5 T
Mean poloidal field $\bar{\bar{B}}_p \equiv I_p \big/ 5\sqrt{ab}$	$\lesssim$ 3.4 T
Poloidal current I_θ	$\lesssim$ 9 MA
Edge safety factor q_ψ	3.6
Confinement strength $S_c \equiv I_p \bar{\bar{B}}_p$	38 $\mathrm{MA \cdot T}$
Plasma volume	$\simeq 10\,\mathrm{m}^3$
Plasma surface	$\simeq 34\,\mathrm{m}^2$
ICRH heating ($\simeq 100 - 140\,\mathrm{MHz}$)	$\lesssim$ 20 MW
Optimal ICRH Heating (115 MHz)	3-5 MW

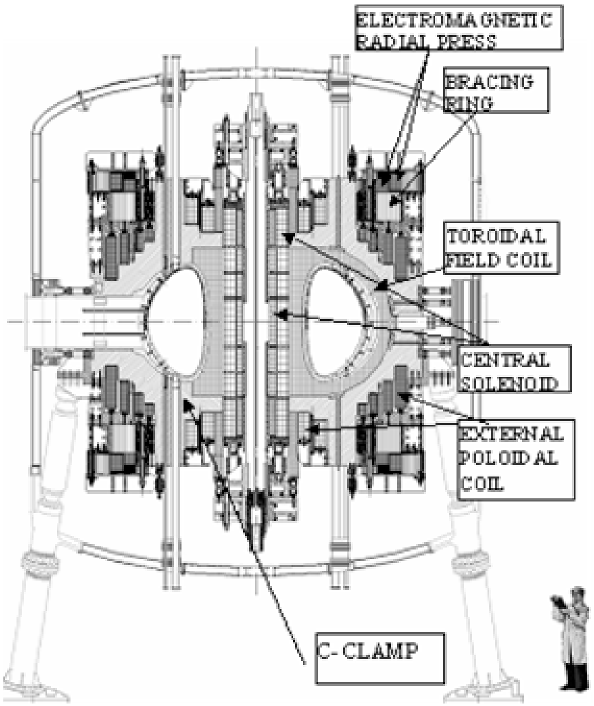
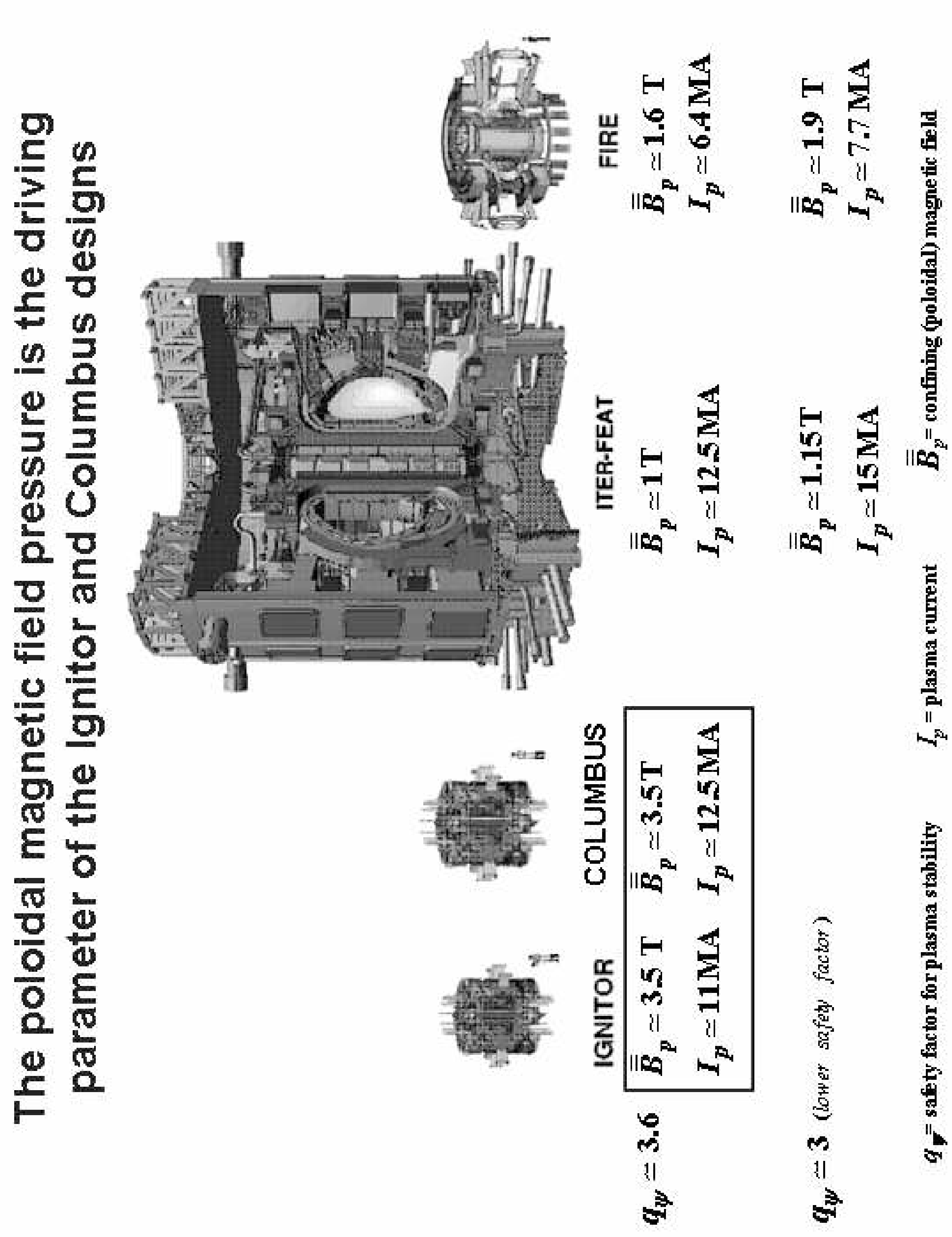
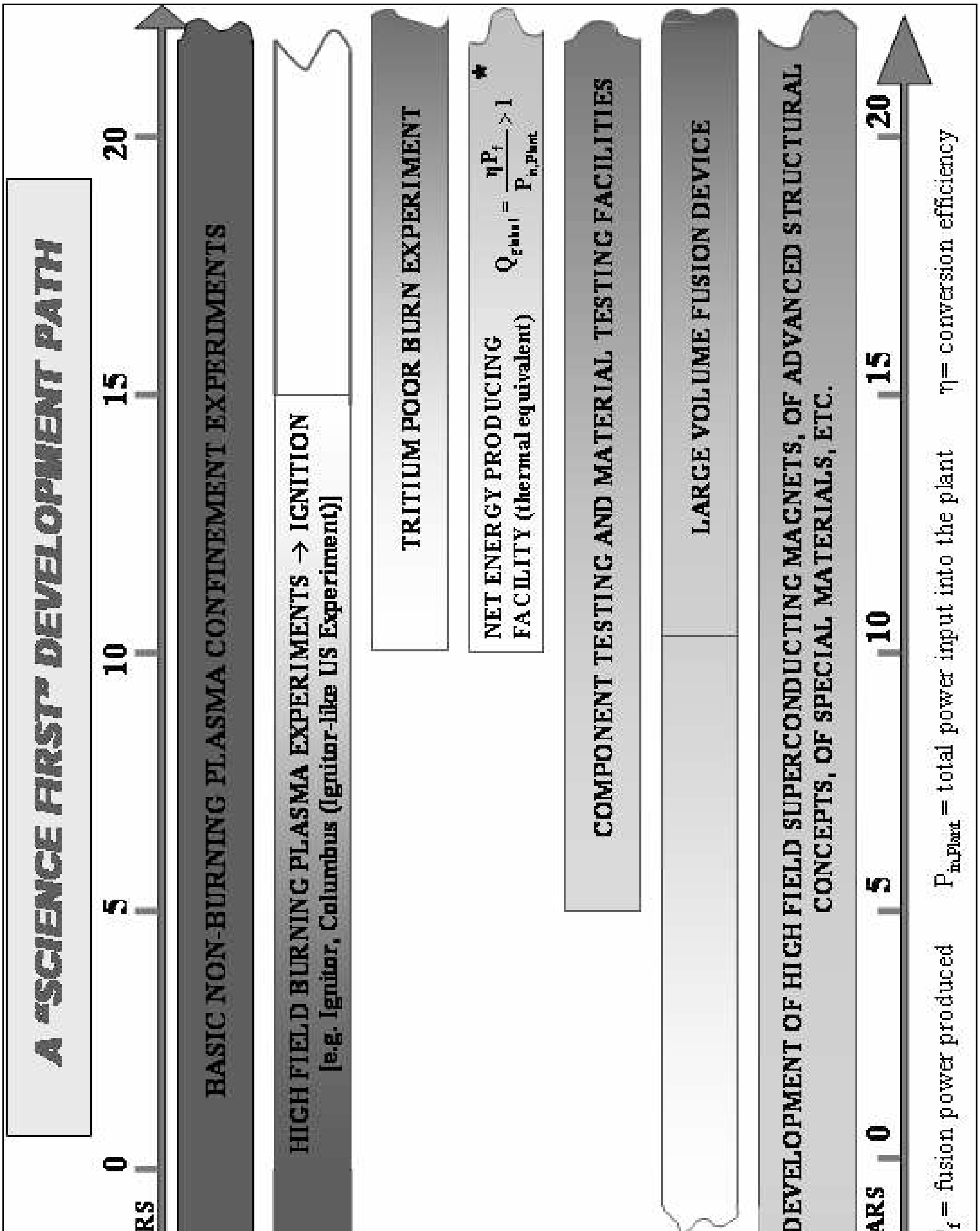

ELECTROMAGNETIC
RADIAL PRESS
BRACING
RING
TOROIDAL
FIELD COIL
CENTRAL
SOLENOID
EXTERNAL
POLOIDAL
COIL
C- CLAMP

Fig. 2

Appendix B: Presently proposed fusion burning plasma experiments



Appendix C: Development path for fusion on research



Appendix D: Relevant Literature for the Ignitor Program

[I] Coppi, B., Sugiyama, L.E., Nassi, M., "Plasma Characteristics for a Compact D-T Ignition Experiment", Fusion Technology, Vol. 21, p. 1612, May 1992.

[II] Coppi, B., Nassi, M. and the Ignitor construction Group, "Characteristics of the Ignitor Ult Experiment", M.I.T. Research Laboratory of Electronics Report, PTP-92/8, July 1992.

[III] Coppi, A.C., Coppi, B., "Stability of internal Global modes in Advanced Plasma Confinement Configurations", M.I.T. Report, RLE-88/17, Cambridge, MA,1989.

[IV] Boxman, G.J., Coppi, B., Dekoch, L.C., et al., "Low and High Density Operation of Alcator", Proc. 7th Eur. Conf. on plasma Physics, Lausanne, 1976 (Ecole Polytechnique Federale de Lausanne, Switzerland) Vol.2, p.14 (1976).

[V] Coppi, B., Detragiache, P., Kulsrud, R., Pegoraro, F., and Tang, W.M., "Physics of Plasmas Close to the Lawson Limit and Under Fusion Burn Conditions," Plasma Physics and Controlled Nuclear Fusion Research, London, 1984 (I.A.E.A., Vienna), Vol.2, p.93 (1985); Coppi, B., and Tang, W.M., "Ignition Density Bounds and Collective Modes of fusing Nuclei", Phys. Fluids **31** 2683 (1988).

[VI] Coppi, B., Cowley, S., Detragiache, P., et al., "Plasma Physics and Controlled Nuclear Fusion Research 1984", Proceedings of the 10th Int. Conf. London, vol.2, pp. 93, 1995 and Coppi, B., Tang, W.M., Phys. Fluids, vol. 31, pp.2683, 1988.

[VII] Shultz, J.H. et al., "Advanced magnets and implications for BPX", BPS Workshop II, G.A., San Diego, 2001.

[VIII] Coppi, B., Englade, E.,Nassi, M., et al., "Current Density Transport, Confinement and Fusion Burn Conditions", paper CN-53/D-4-14 presented at the 13th IAEA Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research, 1990, Washington D.C., USA.

[IX] Rollet, S., Pedretti, E., "Exact and Approximate Geometric Calculations for Toroidal D-Shaped Plasmas: Application to Ignitor-Ult", Associazione EURATOM-ENEA sulla Fusione, Dipartimento di Energia, RT/ERG/FUS/94/12, Roma, 1994.

[X] Sigmar, D.J., Bateman, G., Bell, M.G., et al., Plasma Physics and controlled Nuclear Fusion Research, (Publ. I.A.E.A., Vienna, (1991), Paper CN-53/G-1-1, (1990).

[XI] Coppi, B., Nassi, M., "Comments on the Ignitor and the BPX (CIT) Concepts". MIT Report PTP-91/05, April 1991.

[XII] Coppi, B., et al., "The Ignitor Experiment", M.I.T. Research Laboratory of Electronics Report, PTP02/5 (2002).

[XIII] L. Bromberg, P. Titus, D. Cohn, C. Bolton, "IGNITOR Scale-up studies(DIGNITOR)", MIT Report PFC/RR-90-5, April 1990.

[XIV] Thome, R.J., Tarrh, J.M., "MHD and Fusion Magnets, Field and Force Design Concepts", Wiley Interscience, 1982.

10. Stresses in Columbus

The scale-up of Columbus consists of increasing the linear dimensions of Ignitor by a factor of 25/22 while maintaining a fixed value for the current within the TF coils' turns. Such conditions allow maintaining the non-thermal stresses within the structures almost unaltered. Conversely, the pulse times, which are limited by the increase of the TF coils temperature, increase according to the square of the linear scale-change. Thus, an increase of the machine dimensions results in the capability of running longer pulses without increasing the stresses within the structure. Many studies carried out in the past confirm these scaling-up criteria. One of them is the study of DIGNITOR, a scale-up of the Ignitor device [17].

Stresses generated by electro magnetic loads

Preliminary considerations on the scaling of the stresses of Columbus can be ascertained by computing the loads in an approximated form. The array consisting of 24 TF coils is approximated to an ideal continuous thin-shell torus having a current sheet uniformly distributed on the surface. The cross-section of the torus is shaped to ensure a constant tension distribution, e.g., the absence of moments. The condition required to have a momentless shape is ρ, the radius of curvature of the coil, to be proportional to r, the local distance of the coil from the z-axis of symmetry [18]. In this configuration, the magnetic load *dF* on the generic infinitesimal coil segment is always normal to the coil and non-uniform because of its proportionality to 1/r. The corresponding total z-directed force on the top half of the TF coils set is

$$N_c F_z = \frac{B_m^2 \pi r_1^2}{\mu_0} \ln\left(\frac{r_2}{r_1}\right) \qquad (7.1)$$

where N_c is the number of coils making the torus, F_z is the z-directed force per half coil, B_m is the maximum magnetic field within the current sheet and r_1 and r_2 are respectively the minimum and maximum distances from the z axis of symmetry to the current sheet. Equation (7.1) depicts the independence of the force F_z from the shape of the cross-section and its dependence on the radial coordinates of the cross section at the midplane. The validity of Eq. (7.1) can be rigorously demonstrated for all the shapes that characterize the family of constant tension cross sections [18]. Cleary, this formulation is not rigorous in the case of Columbus and Ignitor where the shapes of the TF coils lead to non-uniform tensions and the distribution is discrete. Nevertheless, this procedure is to quantify the magnitude of the forces acting on the TF coils with an approximation within 10-15%. In the case of Ignitor, Eq. (7.1) produces a total vertical force, at the worst conditions, of about 780 MN on the top half of the TF magnet system and it compares to a value of about 820 MN obtained from detailed finite element analyses. According to Eq. (7.1), the force on Columbus is about 945 MN and it scales with the square of the increase in linear dimensions (χ = 25/22) and the square of the decrease in the toroidal field (ζ= 0.968). Introducing the suffix *Col* to describe all the values relative to Columbus and the suffix *Ign* to describe those relative to Ignitor, defining χ as 25/22, e.g. the increase in linear dimensions of Columbus relative to Ignitor, and accepting that

$$r_{1,Col} \approx r_{1,Ign}\,\chi\,,$$

$$r_{2,Col} \approx r_{2,Ign}\,\chi\,,$$

$$B_{m,\,Col} \approx \zeta\, B_{m,\,Ign}$$

we have $F_{z,Col} \approx F_{z,\,Ign}\,\zeta^2\chi^2$. It is straightforward to conclude that the normal stresses σ vary within the TF coils proportionally to the square of ζ, being

$$\sigma_{z,Col} = \frac{F_{z,Col}}{S_{Col}} \cong \frac{F_{z,Ign}\zeta^2\chi^2}{S_{Ign}\chi^2} = \zeta^2\sigma_{z,Ign} \qquad (7.2)$$

where S is the surface on which F_z is applied.

Stresses generated by thermal loads

The scaling of the thermal stresses within the TF coils is more complicated relative to that of the electro-magnetic stresses. Ohmic heat and nuclear heat are the two sources that contribute to the increase of thermal stresses. Ohmic heat has a predominant effect on the TF magnets' temperatures compared to nuclear heat. This is due to the presence of the First Wall and the plasma chamber that partially shield the TF coils from the neutron power deposition and, furthermore, to the limited duration in time of the nuclear loads.

I) Ohmic heat: A preliminary finite element analysis has been carried out to study the temperature distribution within the TF magnets of Columbus during a reference plasma pulse. First, an electro-magnetic simulation was run to determine the time-dependent current distribution within a single magnet. Then, a thermal analysis was performed to determine the consequential temperature distribution. The simulations confirmed the results of simplified calculations done by hand, which suggest an inverse proportionality between the length of the plasma pulse and the square of the current density flowing within the TF coils

$$\frac{t_{Col}}{t_{Ign}} \approx \left(\frac{j_{Ign}}{j_{Col}}\right)^2 \qquad (7.3)$$

where t is the time variable and j is the average value of the current density.

If the contribution of the nuclear heat is neglected, the maximum temperature reached in the Ignitor TF coils is about 187 K at the plasma side of the inner leg. This temperature is reached at the end of the plasma pulse, when the coil stores the maximum energy and the thermal conduction hasn't flattened the temperature profile within the copper plates yet. An analogous FE analysis was run introducing the parameters of the Columbus TF coils. The length of the plasma pulse was extended by a factor $N^{-2}\chi^4 \simeq 1.38$ (ref. Table I) and the presence of the nuclear heat disregarded. The maximum temperature value, 181 K at the inner leg of the coil, is fairly consistent with the value estimated by using the scaling criteria (7.3).

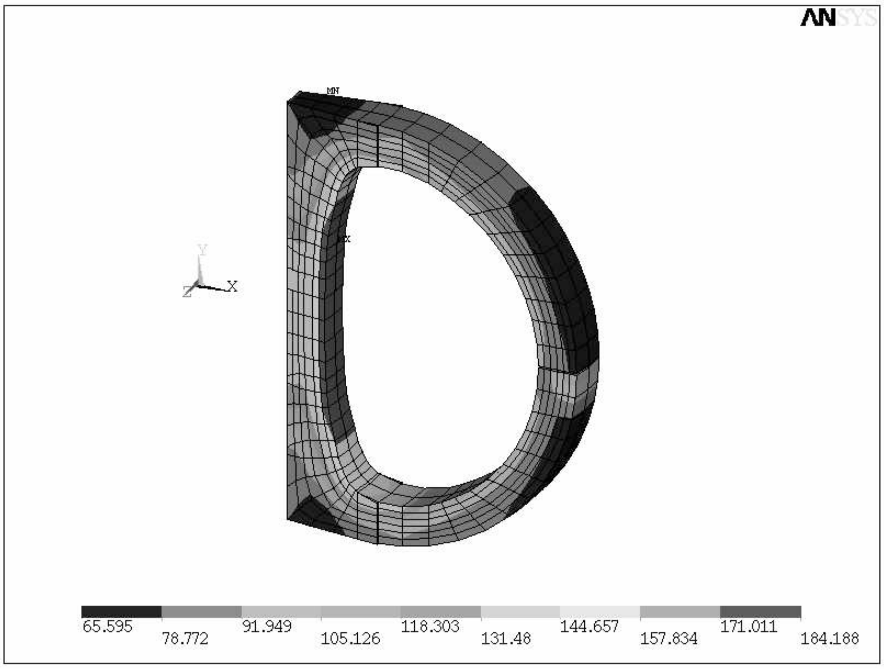


Figure 1: Maximum temperature distribution [K] obtained at the end of the reference plasma pulse and assuming a flattop that lasts 5.6 s. (No neutron thermal loads have been considered).

In Columbus, as in Ignitor, the current flow is for the greater part distributed within a limited region of the turns' cross-section. The current flow tends to follow the shortest electrical circuit and it concentrates toward the plasma cavity side of the TF coils. Thus, the increased dimensions of the Columbus device relative to Ignitor do not provide an important amelioration of the ohmic heat distribution within the turns of the TF magnet coils. A subdivision of each TF coil in two separated nested coils would guarantee extremely improved operations, but the technical complexities involved and the increase in cost of the system make a traditional toroidal field configuration still preferable.

II) Nuclear heat: Thermal stresses generated by the nuclear heat within the TF coils increase relative to those of Ignitor. In fact, the amount of nuclear heat scales to the volume of the plasma ($\propto \chi^3$)

while, since the e-folding neutron absorption distance[2] of metals remains fixed to about 9-10 cm, the volume on which the thermal energy dissipates scales only to the square of the linear dimensions ($\propto \chi^2$). The heat capacity of copper being a growing function of temperature, the increase of temperature ascribed to the merely nuclear heat is dominant at lower temperature locations. Simple hand calculations, based on the relation

$$E_{tot} = \rho \cdot c(T) \cdot \Delta T \qquad (7.4)$$

and on a nuclear heat value of 20 MW/m^3, suggest a temperature growth less than 70 K at the colder locations of the TF magnet and less than 35 K at the plasma side of the inner leg, where the maximum temperatures are normally experienced. The preliminary analyses performed so far suggest that the temperatures reached in the Columbus TF magnets are well below the limit of 240 K chosen to ensure a high structural reliability.

[2] Limited variability is introduced by the difference in density of metals.

11. Brief description of the auxiliary systems

1. Tritium System

Tritium required for the D-T discharges in Columbus. The limited number of discharges in these conditions keeps the inventory of tritium at levels that are comparable to those of Ignitor. Thus, the tritium system of Columbus will be designed similarly to the one of Ignitor [reference].

2. Remote Handling

The activation of the plasma chamber and magnet materials will require a Remote Handling (RH) system for component maintenance, soon after the beginning of D-T operation. The activation issues mainly concern the First Wall, the Inconel 625 Vacuum Vessel, the Toroidal and Poloidal Field Coils and the AISI 316 LN C-Clamps. Particularly attention will be given to the short-term activation of each component to determine whether or not RH is necessary. The experience gained in analyzing the activation of the components of the Ignitor device will be exploited at a large extent.

The ex-vessel requirements are minimal because of the low neutron fluencies allowed by the optimized design of the machine shielding. The possible maintenance tasks will be required mainly towards the end of the experimental life.

The maintenance work will be located mainly in proximity to the horizontal equatorial ports where a stronger activation is expected. Vertical access through the cryostat can be provided to allow easier ex-vessel maintenance operations. Some of the functions of the in-vessel RH system are necessary for the machine assembly due to human access limitations, and therefore this system is considered as having a higher priority. During the operational period, in-vessel interventions will be carried out with the vacuum vessel cooled and vented. At least one containment barrier will be required at the port openings in order to account for the safe handling of in-vessel components and to prevent the release of hazardous material.

The functions to be performed by the in-vessel RH can be summarized in:

a. the welding operations of the last two sectors of the Vacuum Vessel during the assembly;
b. the installation of the First Wall plates;
c. the installation of the ICRF antennae straps and Faraday shields;
d. the positioning of diagnostic optics;
e. the monitoring of the plasma chamber;
f. the replacement and maintenance of all the above mentioned in-vessel components.

3. Electrical Power Supplies

4. Remarks on Safety Evaluations

12. Engineering requirements

The design of Columbus will utilize many of the solutions adopted for Ignitor and it allows reaching the plasma parameters that ensure the stability of the chosen equilibrium configurations of the plasma column; it is guided by cost-benefit considerations and utilizes well proven technologies to the maximum possible extent. A first estimation based on the Ignitor’s design suggests a five year long construction schedule and an operation period of approximately 10 years.

The device is conceived to:

i) produce and control the desired variety of plasma equilibrium configurations, with plasma currents up to the reference values;

ii) induce the toroidal plasma current and maintain the plasma discharge for an adequate number of energy confinement times;

iii) sustain a well confined plasma column under ignition conditions;

iv) operate with an acceptable thermal wall loading;

v) sustain all the relevant static, dynamic, thermal, electromagnetic and disruptive loads;

vi) have a reasonable cooling down time of the toroidal and poloidal field magnets between discharges;

vii) minimize the electrical power and energy requirements;

viii) assure adequate reliability and durability of internal and external maintenance systems.

Sufficient margins are guaranteed to allow possible optimizations of the plasma performances. Finally, the increased dimensions to Ignitor ensure a higher degree of access for diagnostics, pellet injection, RF antennae, vacuum pumping system, remote maintenance, etc.

13. Reference Full Size Configuration of the Plasma

The plasma configuration of the Columbus device is derived as a scaling of the Ignitor parameters. The machine is based on an axisymmetric confinement configuration designed to produce, under proven conditions of macroscopic stability, high plasma currents and elevated current densities in order to reach the temperature and energy confinement necessary for ignition. This involves the adoption of an elongated cross section, a tight aspect ratio, high magnetic fields and compact dimensions. In this regard, high field, tight aspect ratio configurations have more favorable plasma stability characteristics than lower field ignition experiments, because of their intrinsically low plasma β.

The triangularity δ, the elongation κ and the aspect ratio A, are normally used to identify the geometry of D-shaped plasmas. Once these parameters are determined and the major radius is given, the configuration of the plasma is completely described and the minor radii a and b of the plasma cross-section can be identified. Columbus adopts the same values of triangularity, elongation and aspect ratio as Ignitor and is characterized by a major radius of the plasma R_0 =1.50 m. Thus, being

$$A = \frac{R_0}{a} \qquad (4.1)$$

and

$$\kappa = \frac{b}{a} \; , \qquad (4.2)$$

the minor radii of the machine are a = 0.535 m and b= 0.980 m.

The increase of the Columbus geometrical dimensions relative to those of Ignitor corresponds to an increase of the plasma volume by a factor 1.45 and to an increase of the plasma surface by a factor 1.3. The volume of the plasma column in Columbus is about 14.5 m^3 and its surface about 44 m^2 [10].

The increased dimensions of Columbus enable it to run longer pulses relative to Ignitor. In fact, the rate of the rise in temperature of the TF coils due to ohmic heating decreases with their dimensions, provided that the value of the current driven in each turn of the toroidal field (TF) magnets is kept constant. In addition, preliminary analyses reveal that the effect of prolonged neutron heating on the TF coils does not overly limit the duration of the pulse.

14. Basic considerations for ignition

Columbus, as well as Ignitor, combines high magnetic fields, large plasma currents and high plasma densities to attain fusion burn and ohmic ignition conditions. This enables the device to reach $Q = \infty$, where Q is the ratio of the fusion power to the difference between the total heating power and the increment in the plasma internal energy, under transient conditions.

A 50:50 deuterium-tritium plasma requires a minimum value of the parameter $\mathrm{n_o}\tau_E \simeq 4 \cdot 10^{20}$ sec/m^3 in order to achieve ignition with $T_{eo} \simeq T_{io} \leq 15$ keV, where n_o is the peak plasma electron density, T_{eo} the peak temperature, and τ_E the energy replacement time [2]. Relatively high values of the plasma density, $\mathrm{n_o} \simeq 10^{21}\,\mathrm{m}^{-3}$, then require moderate values of τ_E. Various high field experiments, among them the Alcator C-Mod machine at MIT and the FT/FTU tokamaks at Frascati in Italy, have already proven the achievability of those values and have demonstrated the favorable confinement properties of high density plasmas.

Advantages of high density plasmas

Experimentally, the maximum plasma density n_o that can be supported correlates with the ratio $B_T(R_0)/R_o$, where $B_T(R_0)$ is the toroidal magnetic field at the center of the plasma column, at major radius R_o. Thus, high plasma densities require a high toroidal magnetic field capable of sustaining a high poloidal field B_p and a correspondingly high plasma current I_p. The values of n_o may be also associated with the average toroidal current density $\langle J_\Phi \rangle$, whose maximum reference value in Columbus is about 7.6 MA/m^2. Experimental results suggest that this value should offer a considerable margin of safety to attain the desired peak plasma density [3].

The combination of large plasma density and efficient ohmic heating allows ignition at relatively low plasma temperatures and, consequently, a reduction of the fusion power and the thermal loads on the walls. An increased plasma density improves the plasma purity, maintains high the concentration of fusing nuclei and reduces the power lost due to Bremsstrahlung radiation. In particular, the parameter $Z_{eff} = \sum_i n_i Z_i^2 / n_e$, where $n_e = \sum_i n_i$ is a measure of the average charge of the plasma nuclei, should not be higher than about 1.6. The most reliable and proven way to keep Z_{eff} below this value, according to the experiment performed so far, is to produce plasmas with high densities. A long series of experiments has confirmed the observation made first by the Alcator machine in late 1974 that the effective charge Z_{eff} decreases monotonically with the density n_e [5] and that the plasma purity is favored by high magnetic field values.

Relatively high plasma edge densities also contribute to confine impurities to the scrape off layer, where the induced radiation helps to distribute the thermal wall loading more uniformly over the plasma chamber surface. The low ignition temperatures associated with high density further help in keeping the plasma clean by reducing the thermal wall loading.

Finally, in Columbus, as well as in Ignitor, the peaked plasma density profiles, if necessary, can be maintained by external means such as a pellet injector. Peaked density profiles stabilize the long length wave η_i modes [6], minimize their contribution to the ion thermal transport [7], and may also act to suppress sawtooth oscillations. Relative high edge density should imply high probability of interactions between plasma particles, neutrals and also impurities. As a consequence, a high ionization rate of neutrals, a high level of recycling, an effective screening of the main plasma from impurities, a low Z_{eff} and a strong radiative cooling at the edge should be expected.

Plasma current

High values of B_p produce a strong rate of ohmic heating, while large currents I_p tightly confine the fast α-particles produced by the fusion reactions, so that they deposit their energy at the center of the plasma column. Furthermore, high values I_p can limit the degradation of the energy confinement time τ_E in the so-called L-regime, which is observed in present-day experiments when an injected form of heating is applied and prevails over ohmic heating. The high value of I_p is an important feature of Columbus, as the value of the parameter τ_E is generally more difficult to predict than the attainable peak density. In addition to the toroidal current I_p, the Columbus plasma configuration, featuring a small value of the parameter β_p ($\beta_{\mathrm{p}} = 8\pi\langle p\rangle / B_p^2$ where $\langle p\rangle$ is the mean plasma pressure), an elongated plasma cross section and a tight aspect ratio, also ensures the presence of a considerable paramagnetic plasma current I_θ flowing in the poloidal direction.

A significant vertical elongation, e.g. $\kappa \simeq 1.8$ in Columbus and Ignitor, substantially increases the plasma current at constant values of B_T and R. In addition, in Columbus the poloidal plasma beta β_p can be kept small at ignition, to improve the plasma stability and, in particular, to stabilize the ideal MHD modes with mode numbers m = 1, n = 1 that are associated with sawtooth oscillations [4].

15. Rationale of the experiment

Columbus strategy to reach ignition

The rate at which degradation of τ_E might take place when the alpha power P_α prevails over the ohmic power P_{OH} is the key factor that underlies the Columbus strategy to reach ignition. Similarly to Ignitor, Columbus has the favorable feature of reaching ignition where P_α compensates for all forms of energy loss. In fact:

i) The degradation of τ_E has been observed so far when ohmic heating becomes much smaller than other forms of heating, all of which are injected at discrete points around the torus. On the other hand, α-heating is internal to the plasma and distributed axisymmetrically, two features that it has in common with ohmic heating, which has optimal confinement characteristics;

ii) In order to preserve a good margin for τ_E, the best strategy is to maintain a strong rate of ohmic heating up to relatively high temperatures when the α-particle heating also begins to be strong. This strategy can be accomplished by programming the rise of I_p and n_e while gradually increasing the cross section of the plasma column [9].

X-points configurations

Columbus is characterized by an optimized set of poloidal coils. They are placed in proximity to the plasma column and enable the machine to generate plasma equilibrium configurations with x-points of the same type as those produced by divertors. These configurations can be obtained by avoiding the presence of narrow regions of the First Wall where the thermal loading would be too high, by replacing the molybdenum tiles with tungsten tiles at specific locations and by keeping I_p well below its maximum design value. Nevertheless, the larger dimensions of Columbus relativeto the Ignitor machine introduce some drawbacks:

6. the requisite of longer time for the increase of all the currents;
7. larger energy stored in the magnetic systems;
8. larger weight of the components;
9. larger dimensions, powers, and costs of the auxiliary systems (electrical power supply, cryogenics, tritium supply and gas fueling, vacuum pumping and plasma exhaust treatments, shielding);
10. larger tritium inventory.

16. Advantages of the Columbus Limiter Configuration

The use of a divertor in tokamaks was first proposed to improve the thermal energy confinement in plasmas before the favorable characteristics of high density plasmas were discovered [11]. At present day, many reasons suggest that a limiter configuration is more convenient relative to a divertor solution:

i) Regimes with a high degree of purity have been obtained in high density plasmas, where the effectiveness of divertors to obtain low values of Z_{eff} has not been demonstrated yet;

ii) The design of the plasma chamber and of the toroidal magnet would become considerably more complex and costly;

iii) The major radius of the machine would have to undergo a large increase, and the attainable values of B/R would be considerably degraded. This degradation would undermine the margin by which the peak densities n_o could be predicted to be produced and well-confined;

iv) The increased dimensions and lower magnetic fields would, assuming that the maximum plasma current can be held at 12.2 MA, lead to lower values of the poloidal field B_p and to a decrease of the maximum temperature achievable by ohmic heating alone, as well as of the maximum plasma pressure that can be confined without driving macroscopic (ideal MHD) internal modes unstable;

v) With the demise of ohmic heating, a large and reliable injected heating system would become necessary, rather than being a back up system. In fact, degraded confinement issues become prominent well before the α-particle heating begins to be a key component of the overall energy balance;

vi) Operations in the divertor mode introduce relatively narrow regions (the divertor plates) where the wall heating reaches very high values [12]. On the other hand, a series of experiments carried out by the FT and FTU machines in high density plasmas and with limiter configurations have shown that a high fraction of the plasma thermal energy (up to 85%) is carried to the First Wall by radiation. Therefore, the peak values of the thermal wall loading can be expected to be modest especially when ignition at temperatures below 15 keV can be achieved;

vii) The density of the Scrape Off Layer (SOL) in a divertor configuration operating in H-mode will be lower and the SOL width will be reduced compared to L-mode operation. These will yield a larger recycle of neutral particles and lower screening effect on impurities.

All these observations [13] suggest the difficulty to reach ignition in the presence of a divertor configuration. Furthermore, all the advantages in terms of compact nature of the experiment, limit of cost, time scales, etc, would be lost. Taking into account that its ohmic operation avoids the degradation of the energy confinement observed in connection with the application of auxiliary heating in present-day experiments, Columbus can reach ignition without the use of a divertor configuration in order to operate in the H-mode of confinement. Similarly to Ignitor [14], the Columbus machine will operate with limiter configurations and without x-points at the maximum plasma currents. The First Wall, i.e. a set of molybdenum tiles that cover the entire inner surface of the plasma chamber, works, in principle, as an extended limiter. The large contact area, about 44 m^2 in case of limiter configurations filling the entire cavity, ensures the heat loads to be widely spread on the surfaces. ~~At the present day, no direct cooling system is considered necessary. Cooling takes places primarily by conduction to the plasma chamber and by radiation.~~

~~The choice of a limiter instead of a classic divertor with coils inserted inside the toroidal field magnets is motivated by the complexities involved in operating a divertor inside a high magnetic field environment. Furthermore, the easier accessibility to H-modes ensured by a divertor does not compensate for the degradation of global plasma parameters (e.g., the maximum achievable current~~ ~~I_p) [14].~~

Table I: Reference Parameters of Columbus compared to Ignitor

Parameter	Columbus	Ignitor
Major radius R_0	1.50 m	1.32 m
Minor radii $a \times b$	0.535 m × 0.98 m	0.47 m × 0.86 m
Aspect ratio A	2.8	2.8
Elongation κ	1.83	1.83
Triangularity δ	0.4	0.4
Vacuum Toroidal Field B_T at $R = R_0$	$\lesssim$ 12.6 T	$\lesssim$ 13T
Toroidal Current I_p	$\lesssim$ 12.2 MA	$\lesssim$ 11 MA
Poloidal Current I_θ	$\lesssim$ 10 MA	$\lesssim$ 9 MA

Paramagnetic Field Produced by I_θ	$\simeq$ 1.4 T	$\simeq$ 1.4 T
Mean Poloidal Field $\bar{\bar{B}}_p \equiv I_p / \left(5\sqrt{ab}\right)$	$\simeq$ 3.4 T	$\simeq$ 3.4 T
Confinement Strength $S_c \equiv \bar{\bar{B}}_p I_p$	$\lesssim$ 41.5 MN/m	$\lesssim$ 38 MN/m
Toroidal Current Density $<J_\Phi> \equiv I_p / \left(\pi ab\right)$	$\lesssim$ 7.4 MA/m^2	$\lesssim$ 9.3 MA/m^2
Maximum Poloidal Field B_{pM} $(R< R_0)$	$\lesssim$ 6.5 T	$\lesssim$ 6.5 T
Edge Magnetic Safety Factor q_Ψ	3.6 @ $I_p \simeq$ 12.2 MA	3.6 @ $I_p \simeq$ 11 MA
Magnetic Flux Swing	$\lesssim$ 37.5 Vs	$\lesssim$ 33 Vs
Plasma Volume V_0	$\simeq$ 14.5 m^3	$\simeq$ 10 m^3
Plasma Surface S_0	$\simeq$ 44 m^2	$\simeq$ 34 m^2

Table I: Scale up criteria.

Parameter	**Scale up criteria**	**Ratio (χ =25/22; ζ = 0.968)**
Electric current density in TF coils	$j \propto N \chi^{-2}$	$j^{Col}/j^{Ign} \propto 0.852$ *
Total electric current	$I \propto N$	$I^{Col}/I^{Ign} \propto 1.100$
Magnet mechanical stress	$\sigma \propto \zeta^2$	$\sigma^{Col}/\sigma^{Ign} \propto 0.937$
Electrical resistivity	$\eta \propto \chi^0$???	$\eta^{Col}/\eta^{Ign} \propto 1.000$
Electrical resistance	$\Omega \propto N^2 \chi^{-1}$	$\Omega^{Col}/\Omega^{Ign} \propto 1.06$
Inductance	$L \propto \chi$???	$L^{Col}/L^{ig} \propto 1.136$
Time constant	$t_e = L\,\Omega^{-1} \propto \chi^2$??	$t_e/ t_e^{ig} \propto 1.291$
Pulse flattop	$t_{ft} \equiv t_e \propto \chi^2$????	$t_{ft}^{Col}/t_{ft}^{Ig} \propto 1.291$

Magnet heating time constant with no cooling	$t_m \propto N^{-2}\chi^4$	$t_m^{Col}/t_m^{Ign} \propto 1.38$
Transformer flux swing	$\Phi \propto \chi$	$\Phi^{Col}/\Phi^{Ign} \propto 1.136$
Inductive flux consumption	$LI \propto \zeta\chi^2$????	$(LI)/(LI)^{ig} \propto 1.250$
Resistive flux consumption	$\Omega/t_{ft} \propto \zeta\chi^2$????	$(\Omega/t_{ft})/(\Omega/t_{ft})^{ig} \propto 1.250$
Neutron power wall loading	$P_W \propto \zeta^4\chi$????	$P_W/P_W^{ig} \propto 0.998$

* Col = Columbus; Ign = Ignitor.

FIGURE 4: COLUMBUS & IGNITOR

FIGURE 5: PORTFOLIO