CHARACTERISTICS OF THE IGNITOR ULT EXPERIMENT

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ABSTRACT

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 reviewed. The concptual foundations of the experiment, based on present-day experimental and theoretical understanding of the physics of magnetically confined plasmas are analyzed. Results of free boundary transport simulations are reported to highlight the operating conditions that must be considered in the engineering analysis. The main engineering solutions and design characteristics of the machine's principal components are presented.

I. INTRODUCTION

The physics underlying the Ignitor experiment [1-7], as well as its engineering characteristics [8-13], have been studied extensively in the last few years. This work aims to review and update the relevant literature.

Since its original proposal [1] in 1975, the Ignitor experiment has undergone a process of continuing development of its engineering design and its plasma parameters, following the results of the experiments that have been performed and the progress made in understanding the physics of high energy plasmas. The goals of the Ignitor experiment have remained unchanged:

- to investigate the collective modes and the transport processes that occur in D–T fusion burning plasmas;
- to attain fusion burning 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 minimizing the reliance on injected heating system. (Here n_o is the peak plasma density and τ_E is the energy replacement time);
- to study the effectiveness of a suitable ion cyclotron radio frequency (ICRF) heating system in heating the plasma to burning 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 tokamak for approaching D–^{3}He burning conditions;
- to test diagnostic systems for burning plasmas.
- to develop methods for the control, heating and fueling of high density plasmas.

II. 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, that had been

discovered by high field machines such as Alcator/Alcator-C at MIT [14] and FT/FTU in Frascati, the TFTR experiments of Princeton and that have been confirmed by other advanced experiments. In fact high field experiments with tight aspect ratio were the first [1] to be proposed that appeared likely to achieve fusion ignition conditions on the basis of existing technology and the known propierties of high density plasmas.
The reference plasma dimensions and parameters of Ignitor Ult are reported in Table I.

TABLE I: Reference Design Parameters of the Ignitor Ult Configuration

$R_o \simeq 1.32$ m	Major radius of the plasma column
$a \times b \simeq 0.47 \times 0.87\mathrm{m}^2$	Minor radii of the plasma cross section
$\delta_G \simeq 0.36$	Triangularity of the plasma cross section
$I_p \lesssim 12$ MA	Plasma current in the toroidal direction
$I_\theta \lesssim 9$ MA	Plasma current in the poloidal direction
$B_T \lesssim 13$ T	Vacuum toroidal field at R_o
$\Delta B_T \lesssim 1.5$ T	Paramagnetic (additional) field produced by I_θ
$\langle J_\phi \rangle \lesssim 9.3$ 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 magnetic safety factor at $I_p = 12$MA
$V_o \simeq 9.5$ m^3	Plasma volume
$S_o \simeq 36$ m^2	Plasma surface area
$P_J \lesssim 16$ MW	Injected heating power (ICRF at $f \simeq 130$ MHz)

These parameters have been chosen in order to obtain:

- a high peak plasma density ($n_o \simeq 10^{21}$ m^{-3}). The maximum plasma density that can be supported has been observed to correlate roughly with the ratio B_T/R_o. On the basis of the Alcator C machine, where $n_o \simeq 2 \times 10^{21}$ m^{-3} was achieved with $B_T \simeq 12.5$ T and $R_o \simeq 0.64$ m, and the TFTR machine at Princeton, where even larger ratios of $n_o R_o / B_T$ were achieved, a configuration with $R_o \simeq 1.3$ m and $B_T \simeq 13$ T should be able to reliably sustain densities of 10^{21} m^{-3}. Furthermore, if the density correlates with the volume averaged toroidal current density $\langle J \rangle$, experimental results suggest that the value of $\langle J \rangle$ in Ignitor ($\langle J \rangle \simeq 1$ kA/cm^2) should offer a considerable margin to attain the desired peak plasma density;
- a high mean poloidal magnetic field ($\bar{B}_p \simeq 3.75$ T) and a correspondingly large toroidal plasma current ($I_p \simeq 12$ MA). High poloidal field can be supported by a combination of the strong toroidal magnetic field and a optimized plasma shape.
- low poloidal beta ($\beta_p = 8\pi \langle p \rangle / \bar{B}_p^2 \lesssim 0.13$ at ignition, where $\langle p \rangle$ is the volume averaged plasma pressure);

- a relatively small volume of the region where the magnetic safety factor q is less than one.

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. By the end of this relatively long ($t_r \gtrsim 3$ to 4 s) transient phase, the electric field is strongly inhomogeneous, being small at the center of the plasma column, where the temperature can achieve relatively high values, and maximum at the edge of the plasma column (corresponding to loop voltages $V_\phi \simeq 1$ V). This condition can be maintained beyond ignition.
- ignition at low temperature, $T_o \lesssim 15$ keV.
- limitation of the degradation in the energy replacement time (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 the ohmic heating. The Ignitor 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 axisymmetrically, two features that it shares with ohmic heating that has optimal confinement characteristics. Thus the degree of energy confinement required to attain ignition at low peak temperature ($T_o \lesssim 15$ keV) with values of the confinement parameter $n_o \tau_E \gtrsim 4 \times 10^{20}$ sec/m^3, should be reached with reasonable confidence.
- a relatively high degree of purity that prevents dilution of the reacting nuclei and loss of internal energy from the plasma core by radiation. In practice, the plasma effective charge Z_{eff} should not be higher than about 1.6 in the Ignitor Ult without auxiliary heating. A wide range of experiments performed so far have confirmed that Z_{eff} is a monotonically decreasing function of the plasma density. The high values of B_T and the 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 that can be maintained by external means such as a pellet injector, if necessary. Peaked profiles maintain stability against the so–called η_i modes that enhance the thermal ion transport;
- good confinement of the plasma and the α-particles produced by the fusion reactions in the central part of the plasma column (a current $I_p \simeq 3$ to 4 MA is required to

confine the orbit of the 3.5 Mev α-particles). The considerably larger currents that Ignitor can produce confines 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), flowing in the poloidal direction. This increases the toroidal magnetic field B_T at $R = R_o$ by about 11%.
- a bootstrap current $(I_{BS}) \gtrsim 10\%$ of I_p at ignited conditions. This current reduces slightly the required magnetic flux variation to be produced by the poloidal magnet system.
- a strong margin for stability against ideal and resistive MHD modes, in particular for macroscopic internal $m^o = 1$ [15] modes whose onset could hamper the attainment of ignition [16].

III. PLASMA PERFORMANCES AND STABILITY

A. Plasma Performances

Recent results of free boundary numerical simulations [3-7], using the Tokamak Simulation Code (TSC) [17], have shown that ignition is most effectively achieved soon after the end of the current rise, to take advantage of the favorable conditions obtained by this phase of the discharge in terms of broad toroidal current density profile and, thus, low volume of the region where $q < 1$ (less than 1/10 of the total plasma volume). This situation, combined with the low value of β_p that should stabilize $m^o = 1$ modes, keeps the effects of potential sawtooth oscillations small.

For the maximum Ignitor parameters given in Table I (reference discharge), ignition can be reached [4], after a 3 sec current ramp, at $t \simeq 4.3$ s, $T_o \simeq 11$ keV, $\tau_E \simeq 0.66$ s, $n_{eo} \simeq 1.1 \times 10^{21}\mathrm{m}^{-3}$ and $n_{eo}/\langle n_e \rangle \simeq 2.2$, corresponding to a thermal stored energy $W \simeq 12$ MJ (details of the numerical results and the transport models used are given in [4]). If the peaking of the temperature profile $T_o/\langle T \rangle$ is limited to approximately 3, then $\tau_E \simeq 660$ msec at ignition, when $Z_{eff} \simeq 1.2$. We notice also that the α-heating power P_α, that is by definition equal to the total power losses (P_L) at ignition, is about 18 MW, while $P_{OH} \simeq 9.5$ MW, and this makes the thermal loading of the first wall relatively mild.

During the current ramp, P_{OH} increases continuously while 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 low. After the current ramp ends, P_{OH} falls gradually as the temperature rises, although not as quickly as the central 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.

Data on this reference discharge is shown in Table II at three different times (ST = start of the simulation, FT = beginning of the current flat top, IG = ignition).

TABLE II: Reference Discharge for the Ignitor Ult

	ST	FT	IG	
t	0.2	3.0	4.3	time (sec)
R_o	1.0	1.32	1.32	major radius (m)
a	0.26	0.47	0.47	minor radius (m)
κ	1.07	1.87	1.87	elongation
δ_G	0.08	0.42	0.43	triangularity
$\ell_i/2$	0.47	0.32	0.4	internal inductance
β_p	/	0.08	0.13	poloidal beta
β	0.06	0.8	1.26	toroidal beta (%)
n_{eo}	2.5	11.0	11.0	peak electron density (10^{20} m^{-3})
$n_{\alpha o}$	/	1.5	12.0	peak α-particle density (10^{17} m^{-3})
q_o	1.75	0.83	0.71	central magnetic safety factor
q_ψ	4.2	3.3	3.6	edge magnetic safety factor
$Vol_{q=1}$	/	1.4	5.8	volume inside the $q = 1$ surface (% of the total volume)
I_p	0.95	12.0	12.0	toroidal plasma current (MA)
W	0.07	7.5	11.7	internal energy (MJ)
T_o	1.1	4.0	11.0	peak electron temperature (keV)
τ_E	0.13	0.71	0.66	energy replacement time (sec)
P_{OH}	1.5	13.0	9.5	ohmic power (MW)
P_α	/	2.0	17.8	α-power (MW)
P_B	0.02	3.2	4.1	bremsstrahlung radiation power (MW)
P_{IC}	0.01	0.4	0.5	cyclotron and impurity radiation power (MW)
V-sec	2.0	29.2	31.4	magnetic flux variation (V sec)
I_{BS}	/	0.6	1.0	bootstrap current (MA)

We notice that low temperature ignition is thermally unstable, since the plasma temperature tends to run away in principle, given the temperature dependence of the fusion reactivity. At the same time, it is not difficult to envision intrinsic plasma processes that may limit the temperature excursion. Among the external means we intend to employ are the injection of pellets with compositions and sizes chosen in such a way so as not to depress the plasma temperature to the point where fusion burning is quenched irreparably.

The flat top phase at the maximum parameters $I_p \simeq 12$ MA and $B_T \simeq 13$ T has been designed to last about 1.0 sec for a 3 sec current ramp, followed by a gradual reduction of the current to 8 MA over 4 sec since our analyses indicate that these values of I_p and B_T

are not necessary to sustain the ignited state because the fusion α-power takes care of the power balance. Moreover, during the ramp down of the plasma current, as has been shown experimentally in TFTR [18] and reproduced by transport simulations [6], the current density is reduced only in the external plasma region ($r \geq 2/3\,a$). Thus the confinement properties of the discharge do not deteriorate with respect to the full plasma current case. 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.

A1. Energy Confinement at Ignition

Moderate amounts of auxiliary heating [7], $P_J \sim$10–15 MW, started during the current ramp, allow ignition with energy confinement time of the order of 0.4 sec, while maintaining very small $q = 1$ regions well beyond ignition (Table III). Similarly [7], in the ohmic case, if the requirement of small $q < 1$ region is dropped, either on the basis of the theoretical analysis or by externally stabilizing the sawtooth oscillations, ignition can also occur at these τ_E's and times of $t_I \simeq$ 5–5.5 sec (Table III).

TABLE III: Degraded Conditions and Injected Heating

	Z_{eff}=1.6	Z_{eff}=1.6	Large Thermal Transport	
Heating	Ohmic	Injected	Ohmic	Injected
P_J (MW)	0	5*	0	15†
Z_{eff}	1.6	1.6	1.2	1.2
$\gamma_i^{\ddagger}$	0.5	0.5	1.5	1.5
t_I (sec)	5.3	3.3	5.0**	3.0
β_p	0.14	0.15	0.14	0.19
W (MJ)	12.8	14.0	12.9	16.6
T_o (keV)	13.0	13.2	13.4	15.1
τ_E (msec)	570	555	470	425
P_{OH} (MW)	8.7	8.8	8.1	6.9
P_α (MW)	22.5	25.2	24.3	39.2
P_B (MW)	4.9	5.4	4.2	4.9
P_{IC} (MW)	1.0	1.1	0.5	0.7
$V_{q=1}$ (%)	> 10	2.3	> 10	1.4

* $P_J \simeq 5$ MW for $t > 1.2$ sec.

† $P_J \simeq 5$ MW for $1.2 < t < 1.8$ sec, 10 MW for $1.8 < t < 2.4$ sec, and 15 MW for $t > 2.4$ sec.

‡ Ion thermal diffusion coefficient $\chi_i = \chi_i^{neo} + \gamma_i \chi_e^{nOH}$, where $\chi_e = \chi_e^{OH} + \chi_e^{nOH}$, reference $\gamma_i = 0.5$.

** Subignited; never reaches ignition.

A2. Plasma Density

For a given level of thermal transport and radiation losses (Z_{eff}), there is an optimum density that minimizes the time to reach ignition. A higher density is necessary to compensate for degraded conditions. Higher density, however, accelerates the toroidal current penetration by lowering the electron temperature, and consequently produces larger $q < 1$ region earlier in time. Furthermore, a large n_o may lead to hollow temperature profiles during the current ramp, introducing a delay in the α–power production.

Broadening the density profile [4,7] beyond $n_{eo}/\langle n_e \rangle \simeq 2.2$ while keeping the same value of n_o allows ignition at slightly higher values of τ_E and with a larger region where $q \leq 1$ (see Table IV). Narrowing the density profile produces only small improvements in ignition time and the size of the region where $q \leq 1$, but allows ignition at smaller values of τ_E (see Table IV).

Broad profiles at lower peak density n_o give better ignition characteristics (last case, Table IV).

The effects of increasing n_o during the current ramp or during the flat top have been investigated for different assumed levels of confinement degradation [4,7]. Better results are obtained by limiting n_o during the current ramp and increasing, if needed, its value during the flat top.

TABLE IV: Effects of Different Density Profiles

Density Profile	Narrow	Reference	Broad	Broad*
$n_{eo}/\langle n_e \rangle$	2.9	2.2	1.5	1.5
n_{eo} (10^{20}m^{-3})	11	11	11	8.4
t_I (sec)	4.1	4.3	4.7	4.3
W (MJ)	10.7	11.7	13.4	12.6
T_{eo} (keV)	11.2	11.0	11.1	13.0
β_p	0.12	0.13	0.15	0.15
τ_E (msec)	615	660	705	675
P_{OH} (MW)	8.8	9.5	9.9	9.1
P_α (MW)	17.4	17.8	19.0	18.7
P_B (MW)	3.2	4.1	5.8	4.2
P_{IC} (MW)	0.4	0.5	0.8	0.6
$V_{q=1}$ (%)	4.0	5.8	> 10	4.8†

* Lower density; $n_{eo} = 6.5 \times 10^{20}$m^{-3} at end of ramp ($t = 3$ sec), increasing afterwards.
† Large low shear region for $q \simeq 1$.

A3. Auxiliary Heating

A few megawatts (e.g. 5 MW, see Table III) of injected heating, started during the initial 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 [4,5,7]. High temperatures in the central region, T_o 10 to 15 keV, act to "freeze in" the central current density. In all the Ignitor cases with injected heating, the $q \leq 1$ region can be kept very small until well beyond ignition. The only region with $q \leq 1$ results from the dip in q (that we do not consider realistic) caused by the specific form of the electrical resistivity adopted to take into account the effects of trapped electrons near the magnetic axis.

B. Plasma Stability

Numerical simulation of discharges where no injected heating is present, shows that during the current ramp [5] the $q \leq 1$ region can be kept small or even nonexistent, depending upon the details of the current density distribution near the magnetic axis. After the ramp, the $q \leq 1$ region continues to grow, but up to ignition its volume can be kept below 1/10 of the total. The size of the $q \leq 1$ region also depends on the details of the plasma temperature profile, and thus on the thermal transport. If τ_E does not degrade severely, ignition can be reached rapidly after the end of the current ramp and the $q \leq 1$ region remains small until past ignition. On the other hand, larger transport losses or any other effect that lowers the plasma temperatures in the outer half of the plasma column, increases the rate of penetration of I_p and thus increases the size of the $q \leq 1$ region at a given time.

We note that it is not strictly necessary to keep q_ψ, at the edge of the plasma column, larger than 3 in order to ensure the plasma macroscopic stability, as shown by the results obtained by the JET machine [19]. Nevertheless the machine design makes it possible to maintain $q_\psi \geq 3$ at all times by properly programming the current in the poloidal field coils [5]. In order to avoid internal reconnection activity, provisions are made to avoid the development of hollow or nonmonotonic q profiles in the interior of the plasma. The free boundary simulation has shown [5] that the development of nonmonotonic q (J_ϕ) profiles can be related to the empirically derived JET stability boundaries in (l_i, q_ψ) space [20,21], when a collisional electrical resistivity that includes the effects of trapped electrons is assumed. We consider controlling the monotonicity of the profiles of less importance to the overall plasma stability than controlling the size of the $q \leq 1$ region, since it is easy to avoid the severely hillow q profiles that are associated with the "locked" or quasistationary modes that often lead to disruptions in JET. A further advantage of avoiding nonmonotonic profiles is the prevention of anomalously fast current penetration during the current ramp, which helps to keep the $q \leq 1$ region small.

B1. Sawtooth Oscillations

A high energy particle population, created by an injected heating system such as ICRF, as demonstrated by results obtained by JET [22] and other machines, or by fusion reactions (α–particles), can suppress the onset of sawtooth oscillations [23,24]. Therefore, the Ignitor strategy to eliminate ideal MHD modes (with poloidal mode number $m^o = 1$), is to program $I_p(t)$ in order to maintain a relatively small volume where $q \leq 1$ and to keep β_p well below the threshold value of β_p^{crit} that holds in the absence of high energy particles [25], until substantial D–T burning is produced.

While the ideal MHD modes may be avoided, there remain resistive modes that are also macroscopic in nature, but become more benign as the temperature increases [26]. In particular, there exists a relatively large region in parameter space where these modes are stable. For the parameters typical of Ignitor the stronger of the two relevant resistive modes is found to be stable when $T_o \geq 5$ keV (typically reached soon after the end of the ramp). Also, even if sawteeth are not completely suppressed, they do not prevent ignition if their period increases sufficiently with increasing temperature or if their amplitude remains limited, as expected from the general behavior of $m^o = 1$ resistive instabilities.

Finally, we note that recent experimental evidence [27] show that peaked density profiles enhance the stability against $m^o = 1$ modes. We point out that the possibility to suppress sawtooth oscillations by creating relatively peaked density profiles, in regimes where only ohmic heating is present, has been indicated as an explanation of the sawtooth free plasmas obtained by the ASDEX machine [28]. This explanation corresponds to our prediction, that the "diamagnetic" frequencies , that depend on the plasma pressure gradients, have a stabilizing effect on $m^o = 1$ modes.

B2. Alfvén Gap Modes

Finally we note that the expected Ignitor plasmas ($n_{eo} \gtrsim 10^{21}$ m^{-3} and $T_o \lesssim 15$ keV) are predicted to be stable [29] against low–n° toroidal and elliptical Alfvén eigenmode (TAE and EAE) gap modes. However, both theory and recent neutral beam injection experiments [30] carried out by the DIII–D machine, show that modes of this type with $n^\circ \sim 5$ to 7 can be the most unstable. The theory for thise mode is incomplete. In addition, the instability threshold observed in the experiments is substantially higher than the one predicted by the existing theory.

C. Plasma Scenarios

The planned programs for the plasma current and magnetic field, with the exception of the lowest current scenario, have the following common features:

- the plasma current initially rises to 5 MA during the first 1 or 1.5 sec of the current ramp since experience has shown that such a rate is not excessive, assuming that the current penetration is a diffusive process;
- it rises from 5 MA to the maximum value in further 2 to 3 seconds, at a rate of 2 to 3.5 MA/sec, as suggested by experience and numerical simulations;
- the current in the toroidal field coils (TFCs) goes to 70 MA–turn in the 4 sec before the plasma breakdown at $t = 0$, since this current is needed to buck the central ohmic solenoid (OHS) against the TFCs at $t = 0$. The TFC current follows closely that of the OHS, even during its charging. The charging time cannot be made shorter than 4 sec in order to keep the power drawn from the grid within acceptable levels;
- from $t = 0$ to $t = 1.5$ sec the TFC current is kept constant for engineering reasons (to reduce the overall heating of the TFC). The corresponding B_T is in fact more than adequate to assure the stability of the plasma column. Subsequently B_T is increased linearly to its maximum value simultaneously with the rise of the plasma current. Then it is kept constant as long as the plasma current is kept constant. At the end of the burning phase, it is reduced to zero along with I_p;

These scenarios were determined partly to satisfy engineering considerations (see section XIV).

The planned D–T plasma scenarios are summarized in Table V.

TABLE V: Planned D–T Plasma Scenarios

- maximum parameter regime
  - plasma current $I_p \simeq 12$ MA
  - vacuum toroidal magnetic field, at $R = R_o$, $B_T \simeq 13$ T
  - ramp up time $t_r \simeq 3$ to 4 sec
  - time after ramp up $t_{ar} \simeq 4.5$ sec (I_p is decreased from 12 to 8 MA after a flat top of 0.5 to 1 sec)
  - ramp down time $t_d \simeq 2$ sec (I_p is decreased from 8 to 0 MA)
- high parameter regime
  - $I_p \simeq 11$ MA
  - $B_T \simeq 12$ T
  - $t_r \simeq 3.5$ sec
  - $t_{ar} \simeq 6$ sec (I_p is decreased from 11 to 8 MA after a flat top of 1 sec)
  - $t_d \simeq 2$ sec (I_p is decreased from 8 to 0 MA)
- moderate parameter regime
  - $I_p \simeq 10$ MA
  - $B_T \simeq 11$ T
  - $t_r \simeq 3.5$ sec

- $t_{ar} \simeq 5$ sec (I_p is constant)
- $t_d \simeq 3$ sec

- intermediate parameter regime
  - $I_p \simeq 8$ to 9 MA
  - $B_T \simeq 10$ T
  - $t_r \simeq 3$ sec
  - $t_{ar} \simeq 6$ sec (I_p is constant)
  - $t_d \simeq 2$ sec
- low parameter regime
  - $I_p \simeq 5$ MA
  - $B_T \simeq 5.5$ T
  - $t_r \simeq 2$ sec
  - $t_b \simeq 6$ - 15 sec (I_p is constant)
  - $t_d \simeq 1.5$ sec

In Table VI we present the results for seven different plasma scenarios, with maximum plasma currents of 8, 10 and 12 MA, using different levels of injected heating power and different locations of the magnetic null point (plasma position) at the breakdown (close to the inside or the outside limiter). The main parameters of these scenarios and the corresponding plasma performances (at ignition, or in sub-ignited case, at the time t_I when the maximum plasma temperature is reached) are reported [31]. The transport models used in these simulations, as well as all the other input parameters, are the same as for the reference discharge described above [4] and repeated as case 7 below.

TABLE VI: Different Plasma Scenarios

Case	1	2	3	4	5^a	6	7	
I_p (MA)	8	8	10	10	10	12	12	Plasma current
	out	out	out	out	out	out	in	Location of the null
P_J (MW)	0	10	0	10	0	0	0	Injected heating power
B_T (T)	10	10	11	11	11	13	13	Toroidal field at R_o
B_p (T)	2.7	2.7	3.3	3.3	3.3	3.9	3.9	Poloidal field
B_{vert} (T)	1.2	1.2	1.5	1.5	1.5	1.8	1.8	Vertical field at R_o
t_r (sec)	3.0	3.0	3.5	3.5	3.5	4.0	3.0	Ramp up time for I_p
t_I (sec)	6.0	4.0	5.6	5.0	5.5^b	5.0^b	4.3^b	igniton time
n_{eo} (10^{20}m^{-3})	8.0	8.0	9.0	9.0	9.0^b	10.5^b	10.5^b	Peak electron density
T_o (keV)	6.4	11.4	9.7	17.2	13.2^b	11.8^b	11.0^b	Peak temperature
τ_E (sec)	0.71	0.35	0.53	0.28	0.57^b	0.59^b	0.66^b	Energy replacement time
P_α (MW)	2.3	10.3	10.8	38.0	20.0^b	21.0^b	17.8^b	α-power

P_{OH} (MW)	6.1	4.0	7.1	4.0	5.8^b	8.9^b	9.5^b	Ohmic power
Z_{eff}	1.2	1.2	1.2	1.2	1.2	1.2	1.2	Effective charge
Q^c	2.0	4.0	8.0	14.0	∞^b	∞^b	∞^b	Performance parameter

^{a} all the parameters are the same of case 3 except for the non-ohmic component of the anomalous thermal diffusion coefficient that has been reduced by one third;

^{b} at ignition.

^{c} $Q \equiv P_\alpha/(P_{OH} + P_\alpha + P_J - dW/dt)$ under transient conditions.

IV. ENGINEERING CONSIDERATIONS

The main engineering requirements of the Ignitor machine are to:

- create and control different plasma configurations;
- induce the toroidal plasma current and maintain the plasma discharge for a time $\gtrsim 10\ \tau_E$ at ignition, where τ_E is the estimated energy replacement time;
- operate with an acceptable thermal loading on the first wall;
- withstand the static, dynamic, thermal, electromagnetic, and disruptive loads on all the affected machine components;
- provide access for diagnostics, pellet injector, r.f. antennae, vacuum system, remote maintenance, etc.
- have a reasonably short cooling time between discharges;
- minimize the electrical power and energy consumption;
- assure adequate reliability and durability and existence of internal and external remote maintenance systems;
- maintain a reasonable project time scale and limit the total cost.

In order to deal with some of the functional problems (plasma configurations, induction of the plasma current, etc.) mentioned above, and taking into account the compact dimensions of the machine it has been decided to avoid mixing structural elements with the electrical conductors. Therefore, the necessary mechanical strength has been obtained by [32,33]:

- designing the copper coils in such a way that they themselves support the greatest possible part of the acting forces. This is achieved by:
  - bucking between the central ohmic solenoid and the toroidal field coils. The plasma scenarios of section C have been designed in order to exploit this concept to the utmost;
  - wedging the toroidal field coils;
  - using, where ever possible, a conducting material (GLIDCOP) with enhanced mechanical properties;

- ○ cooling the coils to 30 K between each discharge;
- arranging the main structural elements outside the toroidal field coils in the form of a complete shell made up of meridian segments (the so-called "C–clamps");
- pre-stressing the C–clamps by means of two bracing rings;
- giving an appropriate degree of rigidity to the pre-stressed structure in order to be able to handle the electrodynamic stresses (rigid structure) while allowing enough deformation to cope with the thermal expansion (deformable structure);
- adding a vertical electromagnetic press to handle the remaining electromagnetic forces.

V. POLOIDAL FIELD SYSTEM

A highly optimized set of 14 up-down symmetric poloidal field coils (PFCs) [34], 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 stable against radial and vertical motions.

Copper OFHC (oxygen free high conductivity) has been selected as the material for part of the central solenoid (OHS) in order to minimize the energy thermally dissipated by these coils (its electrical conductivity is 100.2 % IACS at 295 K and 855% IACS at 77 K), while a material with enhanced mechanical properties (GLIDCOP) (dispersion strengthened copper, 0.3% aluminum oxide, ultimate tensile strength 5 MPa (77 K) and 445 MPa (295 K)) is used for the other poloidal field coils that 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, they 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). It has good mechanical properties [shear strength = 104 MPa (77 K) and 48 MPa (295 K)].

The OHS consists of a double array of copper coils with an intermediate of 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 medium is gaseous helium. The initial temperature before a plasma discharge will be lowered to about 30 K. The final temperature should not exceed 260 K in any coil after the long pulse corresponding to the maximum parameters. The cooling time of the OHS after a current pulse corresponding to the maximum plasma current scenario has been evaluated to be about one hour.

The radial outward force acting on the coils of the OHS that are close to the midplane (T1 to T4) is supported in part by bucking against the toroidal magnet coils, while the other coils (T5 and T6) are clamped by a stainless steel cylinder.

A. Poloidal Flux Requirement

An analytical assessment [31] of the magnetic flux variation linked with the plasma column has been carried out and the results have been verified by numerical analyses performed using the TSC code.

The maximum poloidal flux requirement at the magnetic axis ($R \simeq 1.35$ m) is $\simeq 35$ V sec, corresponding to $\simeq 30$ V sec at the plasma edge. This value has been determined by analyzing the different plasma scenarios reported in Table VI under a range of different values of plasma current, ion and electron thermal conductivity, effective charge, plasma density, current ramp time, flat top time, etc.

The volt–second consumption during the flat top, at a rate of about 1.5 V sec per second (for the 12 MA plasma), is due to resistive losses as well as an inductive component corresponding to an increase in the plasma internal inductance. This increase in the internal inductance occurs because the current profile at the start of the flat top has not yet reached the steady state, fully penetrated profile, due to the relative short current ramp compared to the diffusion resistive time scale and the relative high value of plasma current. This feature is useful [4] in order to control the size of the region where the magnetic field line parameter $q \leq 1$ and to avoid the excitation of sawtooth oscillations. It also means that a significant amount of poloidal flux must be provided during the flat top to compensate for the increase in the magnetic energy content of the plasma.

Numerical TSC simulations [7] have shown that programming the plasma density evolution can reduce the dissipative consumption by about $0.7\,\mathrm{V}\cdot\mathrm{s}$. This is accomplished by delaying the increase in the plasma density during the first part of the current ramp, in order to reach a relatively high peak plasma temperature ($T_o \gtrsim 4$ keV) and thus a low plasma resistivity, as soon as possible. Clearly this procedure must allow an adequate current penetration rate in order to satisfy stability criteria and to guarantee a sufficient level of ohmic heating.

Discharges aided by injected heating ($P_J \simeq 10\,\mathrm{MW}$), where the plasma reaches ignition during the current ramp , require up to $\simeq 7$ V sec less than an equivalent ohmic discharge due to the lower value of plasma current, the increase of the bootstrap current fraction, and the higher plasma temperature that reduces the resistive loss.

These values of poloidal magnetic flux variation can be delivered to the plasma by the poloidal field system that is actually designed to produce a flux variation of about 30 V sec at the plasma edge or about 35 V sec at R_o. This last value is limited by thermal and mechanical requirements of the central solenoid.

B. Plasma Equilibria

Several kinds of plasma equilibria can be produced [35]:

- limiter configurations that fill the entire cavity of the plasma chamber are useful to keep the thermal wall loading as uniformly distributed as possible and to minimize the out of plane forces on the toroidal magnet;
- transient double x-point configurations that can be used to reproduce the characteristics of the so-called H–regime where τ_E is only slightly degraded, relative to that expected for regimes where only ohmic heating is present, when other forms of heating prevail. 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 have exceeded the desirable limits, the equilibrium is made to evolve into a limiter configuration;
- detached limiter configurations that enable the plasma column to maintain its reference dimensions and characteristics while having its outer edge detached from the first wall by a distance larger than $a/10$. This should be sufficient to establish the H–regime, following a procedure suggested and confirmed by a significant set of experiments [36].

C. 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 of the plasma cross section and the position of the magnetic axis;
- keep the plasma in the desired position relative to the first 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).

D. Vertical Position Control

Elongated plasma configurations are potentially unstable to vertical motion. An extensive numerical simulation [37] of the plasma dynamics in Ignitor has been carried out using the TSC and the TEQ codes. The stability behavior of the plasma/plasma chamber/PFCs system has been analyzed, taking into account, in a two dimensional space, the real geometry of the machine, the position of the diagnostic magnetic pick up coils and the characteristics of the power supplies in terms of voltage limitations and the response time. The minimum plasma vertical displacement growth time (τ_v), defined as the reciprocal of the linear exponential growth rate of the plasma vertical position, is about 12 ms when the presence of the thick plasma chamber is taken into account and the PFCs are connected in such a way that antisymmetric currents are induced in the up and down symmetric coils. The toroidal magnet coils have an additional beneficial effect in slowing the plasma motion that has not been considered in the present analysis.

The effect of an additional passive structure has been extensively examined:

- the most convenient position to locate a passive coil is in the outboard region, at a poloidal angle $\theta \simeq 75^\circ$ from the midplane;
- the efficiency (ϵ) in slowing the plasma motion drops rapidly as the passive coil is moved away from the plasma (assuming $\epsilon = 100\%$ for a passive conductor located on the plasma chamber, ϵ is $\simeq 20\%$ for a distance of 0.1 m from the plasma chamber);
- a thin layer of copper ($\simeq$ 1 mm) deposited by means of a plasma spray technique on the exterior surface of the plasma chamber (except than for the inboard and outboard region close to the midplane) increases τ_v to about 26 msec. The increase in poloidal flux requirement due to this thin layer of copper has been evaluated to be about 0.7 V sec.

Assuming $\tau_v \simeq$ 26 msec and using the real characteristics of the power supply (maximum voltage $\pm$ 2500 V, time delay in the response 3.8 msec) it is possible to design a feedback system capable of controlling the plasma vertical position.

The best results, in term of fast plasma position control and low power requirement, use a combination of one inboard coil (T4) and one outboard coil (S1 or S2). The inboard coil is less coupled to the plasma and cannot control the plasma motion by itself. However, it produces a radial magnetic field that diffuses rapidly into the plasma and slows the vertical motion, reducing the power requirement on the outboard coil. Both proportional and derivative gains are used in the feedback algorithm for the current in the inboard and outboard coils, while a simple proportional gain is used to determine the voltage in the coils.

Some simulations have been carried out to evaluate the influence on τ_v of small variation around the reference value of some plasma parameters, such as β_p, l_i and κ. The results show that β_p and κ have a very weak effect on the plasma vertical motion. The relatively mild influence of κ (for variation $\Delta\kappa = \pm 5\%$) on τ_v is due, as explained in ref.[38], to the presence of a close-fitting vacuum vessel. If κ is increased, a larger field index ($n_v \equiv -\frac{R_o}{B_z}\frac{\partial B_z}{\partial x}$) is required for the equilibrium and the plasma becomes more unstable. However, a more elongated plasma also lies closer to the plasma chamber and experiences a stronger stabilizing effect. Larger values of l_i give rise to lower values of τ_v, as already found [38], because for narrow toroidal plasma current distribution (large l_i), the plasma current lies on average farther away from the plasma chamber. Futhermore the equilibrium configuration at large l_i requires a larger field index for the same elongation.

VI. TOROIDAL FIELD SYSTEM

In order to make Ignitor Ult suitable for relatively long plasma current pulses that can exceed 10 τ_E at ignition, the TFCs have been designed with lower values of the starting temperature (30 K) and of the current density (< 100 MA/m^2) compared to the case of the Alcator-C machine (temperature $\simeq 80$ K and current density up to 220 MA/m^2). The main characteristics [39,40] of the toroidal field system are reported in Table VII.

TABLE VII: Toroidal Field System

- number of coils = 24
- number of turns/coil = 10
- insulation between turns = 0.7 mm
- ground insulation = 2 mm
- packing factor $\simeq 0.9$
- number of cooling circuits per turn = 1 + 1
- number of cooling circuits per coil = 10 + 10
- weight:
  - 1 coil $\simeq$ 4 ton
  - toroidal magnet $\simeq$ 96 ton
  - 1 sector (2 coils + 4 C–clamps) $\simeq$ 28.4 ton
  - 12 sectors $\simeq$ 340.8 ton
- dimensions of the complete assembly (12 sectors):
  - internal diameter = 1.007 m
  - external diameter = 5.5 m
- maximum current density (inboard) $\simeq 108 \times 10^6$ A/m^2
- maximum current density (outboard) $\simeq 72 \times 10^6$ A/m^2
- maximum current $\simeq$ 350 kA
- electrical conductivity $\simeq$ 97.16 % IACS minimum
- conductor material:
  - copper Cu ETP 99.9 (elastic tensile strength $\simeq$ 325 MPa) or
  - copper Cu–Ag 0.1% (elastic tensile strength $\simeq$ 340 MPa)
- insulation material:
  - glass epoxy composite (glass 60 %, resin 40%)
  - resin impregnation by ARALDIT F

A. Structural Solution

A feature maintained throughout the evolution of the design of Ignitor has been that of the TFC 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 [32,41]. In particular, as mentioned before, the loads on the inner leg of the TFC are supported by:

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

As far the out of plane forces are concerned, two main measures have been taken:

- minimize these forces by reducing the crossing angle between polidal field magnetic lines and the current lines of the TFC. This has been implemented, with a good approximation, giving the TFC the same shape as the last closed plasma surface;
- avoiding stress concentrations by using the friction existing in the structure, instead of keys, bolts or tenons, to take the out of plane forces.

B. Cooling System

A hybrid cryogenic system [42,43] is adopted for the cooling of the TFCs, where the warmer part of the plant is operated with liquid N_2 and the colder part with gaseous helium. The conductor's temperature before a plasma discharge is assumed to be about 30 K. After a current pulse, corresponding to the maximum plasma current 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. Cooling of toroidal magnets down to 80 K with the help of liquid N_2 has been done before and the technology is well established. An important advantage of liquid N_2 is its ready availability and easy storage. The step from 80 K to 30 K requires the use of gaseous helium as a cooling and heat transfer medium. 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 the passages. This cooling system has a further advantage: when the coolant temperature at the outlet of the magnet reaches a value below 80 K, the same helium cooling plant can produce an outlet temperature lower than 30 K and shorter the magnet cooling time. The cooling time after a current pulse corresponding to the maximum plasma current scenario has been evaluated to be of the order of 5 hours.

VII. STRUCTURAL COMPONENTS

The main mechanical and electrical sctructural components are [32,41,44]:

- a set of 48 steel plates (AISI 316 LN) or "C–clamps" [45] that surrounds each of the 24 modules of the TFCs. The C–clamps have the following important functions:

    - to contain the horizontal expansion forces, directed outward, of the TFCs. This is possible since the C-clamps are embraced at the extremities by the toroidally continuous bracing rings;
    - to contain most of the vertical expansion force of the TFC. This is possible not only because the C–clamps embrace the TFC, 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 on two protrusions of the C–clamps. In this way the nose of the C-clamp rotate inward, squeezing the inner legs of the TFC. Thus only a small fraction of the vertical separating force is unloaded onto the central leg of the TFC;
    - to stabilize the machine and to 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 OHS and TFCs and the wedging of the inner leg of the TFCs could become unknown and excessive;
    - to support the main machine components: plasma chamber, TFCs, PFCs, etc.;
- two bracing rings 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 tensioning system 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 central post [46] (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 set of heads, two on each side, that are divided into three parts and electrically insulated. The outer head is attached to the fir roots of the post; the inner one transfers the press force to the loading piston;
- a pressure piston, a segmented cylinder, electrically insulated, that transfers the force

of the electromagnetic press without producing any bending moment on the inner leg of the TFC;

- a vertical electromagnetic press (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 TFC becomes significant, or whenever the machine is operated with magnetic fields below the maximum considered values;
- a set of supporting legs, attached to the C–clamps, that support the machine weight and that are designed to take care of seismic effects.

VIII. PLASMA CHAMBER

The plasma chamber [47,48] has been designed in order to produce a structure able to withstand both static and dynamic loads with good reliability, while minimizing the construction and assembly difficulties. The result is a relatively thick chamber made of Inconel 625, divided in 12 sectors that can be assembled and joined by welding. The main characteristics of the plasma chamber are reported in Table VIII.

TABLE VIII: Plasma Chamber

- plasma chamber
  - external radius = 1.813 m
  - internal radius = 0.807 m
  - height = 1.843 m
  - thickness (inboard) = 17 mm
  - thickness (outboard) = 26 mm
  - thickness (join region) = 26 mm
  - thickness (close to the equatorial ports) = 35 mm
- equatorial ports
  - number = 12
  - internal height = 0.8 m
  - internal width = 0.17 m
  - thickness = 18 mm
- equatorial "pocket"
  - number = 12 (6 house the ICRF antennae, the rest are available for the vacuum and the diagnostic systems, etc.)
  - internal height = 0.8 m
  - internal width = 0.5 m
  - internal depth = 0.131 m

    - thickness = 40 to 50 mm
- main vertical ports
    - number = 6
    - dimensions = 35 × 100 mm
- material: Inconel 625
    - ultimate tensile strength ≃ 1400 MPa (77 K), ≃ 850 MPa (295 K)
    - elastic tensile strength ≃ 950 MPa (77 K), ≃ 429 MPa (295 K)

The plasma 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 support has been the ability to withstand both vertical and axisymmetric disruptions with plasma current decay rates from 1 to 2.5 MA/msec. The results of a two-dimensional numerical analysis [49] show that the stresses are below the limit imposed by ASME rules. In the near future, a three-dimensional code will be used to model the toroidal forces due to the discontinuities introduced by the horizontal ports. (This situation has been found [50] to produce significant changes in the force distribution on the plasma chamber in BPX.)

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. Additional external resistive heating is provided to heat the port regions of the chamber.

IX. FIRST WALL SYSTEM

The first wall [51] is made of graphite tiles (18 mm thick) that cover the entire inner surface of the plasma chamber. Thus, in principle, the whole first wall 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 ranges from the total first wall surface ($\simeq$ 36 m^2), for plasma limiter configurations filling the entire cavity, to the area of the inner part ($\simeq$ 12 m^2), for plasmas detached from the outside wall.

A set of 4 tiles, is 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 properties and plasma-related properties such as atomic number, sputtering yield, release and inventory of hydrogen isotopes, radiation damage, the relatively large amount of experimental data available on its behavior as first wall material, etc. 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 a temperature of $\simeq 600$ K.

A. Operating Conditions

The operating conditions of the first wall are the following:

- normal discharge (thermal wall loading $\simeq 1$ MW/m^2):
  - average material temperature $\bar{T}_{surf} \simeq 160$ K;
  - surface material temperature $T_{surf} \simeq 250$ K;
- disruption conditions:
  - surface material temperature $T_{surf} \simeq 3100$ K; the surface reaches the sublimation temperature of the material since the thermal conductivity is not high enough to dissipate the incoming heat flux;
  - the amount of ablated material during a disruption is relatively small (estimated to be 140 - 200 μm);
  - during a vertical disruption, a halo current of the order of 10 to 15 % of the plasma current, corresponding to a halo current density of $\simeq 0.4$ A/mm^2, may be generated. In order to reduce the risk of arcing, the electrical resistance of the first wall - plasma chamber system has been reduced as much as possible.

B. Thermal Wall Loading Analysis

Numerical transport simulation [4,5] have shown that Ignitor can achieve ohmic ignition at low peak temperature ($T_o \gtrsim 11$ keV) with a moderate value of the α-power ($P_\alpha \simeq 18$ MW) (Table II). About 26 % of this power is lost by radiative processes ($P_R \simeq 4.6$ MW) in the main plasma and the remaining 13.4 MW (74 %) is transferred to the scrape off layer (SOL). From here this power is released to the first wall surface by radiative and convective-conductive energy transfer. The importance of one process relative to the other is determined by the plasma behavior in the SOL. Simple physical models [52,53] and experimental data from high density, high magnetic field experiments confirm the expectation of a high density, low temperature plasma in the SOL. This means that the core plasma will be shielded from the impurities, and that this cold and dense scrape–off plasma will be characterized by a strong radiative cooling. In particular, on the basis of an extensive series of observations made on high field machines, it is realistic to consider that at least 50% of the power transferred into the SOL will be released by radiation.

Therefore, we may write:

$$\Gamma_{fw} = \frac{P_L}{S_{fw}}(f_{rad} + f_p f_{con} f_{eff})$$

where Γ_{fw} is the thermal wall loading on the first wall, P_L is the total power loss , S_{fw} is the total area of the first wall, f_{rad} is the fraction of P_L transfers to the first wall by radiation process, f_p is the peaking factor for the convection and conduction processes in the SOL, $f_{con} = 1 - f_{rad}$ is the fraction of P_L transfer to the first wall by convection and conduction processes, and f_{eff} is a geometrical peaking factor that takes into account the effective area of contact between the plasma and the first wall.

For a limiter configuration filling the entire plasma chamber, assuming: $S_{fw} \simeq 36$ m^2, $f_{rad} \simeq 0.6$, $f_p \simeq 3$, $f_{con} \simeq 0.4$ and $f_{eff} \simeq 1$ we may estimate the maximum thermal wall loading to be $\Gamma_{fw} \simeq 0.9$ 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 $\Gamma_{fw} \simeq 2.0$ MW/m^2. Instead, if the plasma interacts only with the inboard surface of the first wall ("detached" limiter configuration) assuming: $S_{fw} \simeq 36$ m^2, $f_{rad} \simeq 0.6$, $f_p \simeq 2$, $f_{con} \simeq 0.4$ and $f_{eff} \simeq 3$, we may estimate the maximum thermal wall loading to be $\Gamma_{fw} \simeq 1.5$ MW/m^2 for $P_L \simeq 18$ MW, and $\Gamma_{fw} \simeq 3.3$ MW/m^2 for $P_L \simeq 40$ MW.

X. AUXILIARY SYSTEMS

A. Injected Heating System

Since ignition can be attained by ohmic heating alone, injected heating systems in compact high field experiments such as Ignitor have the role of backups; to be available, if needed, to suppress the possible onset of sawtooth oscillations, to control the temperature evolution and the current density profiles, and to accelerate the attainment of ignition.

Among the options considered, 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 for Ignitor because of the experimental evidence of its effectiveness in relatively high density plasmas. The antennae are placed in six housings inserted into the vacuum chamber and connected to the large horizontal ports. The power that can be delivered via each housing is in the range of 2.5 to 4.0 MW.

B. Pellet Injector

An injector of deuterium or deuterium-tritium pellets ($\simeq 4$ mm diameter) is considered, in addition to the well-tested technique of gas injection ("puffing"), to create and maintain the desired density profile. Pellet velocities of 2 km/s or higher that are required to reach the central region of the plasma column have, in fact, been already achieved with existing technologies. Another use for a pellet injector that has been recently demonstrated is to condition the first wall by launching lithium pellets into the plasma column prior to regular hydrogenic discharges [54].

C. Tritium Storage and Delivery System

A maximum number of 6 tritium discharges per day has been planned in Ignitor. 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 puffing. There are two injection points and the duration of the gas puffing is 0.1 s.

A complete study [55] of the tritium storage and delivery system (TSDS) has been recently concluded by a joint team from the IFP-CNR of Milan and the Ontario Hydro Research Division. The TSDS has to provide the following function [43]:

- receive tritium as tritide or in a gaseous state;
- check the purity of the tritium and store it as tritide on a uranium storage bed (the maximum quantity of tritium allowed on site is 1 g $\simeq 10^4$ Ci);
- purify and maintain the purity of the stored tritium;
- deliver the required amount of tritium (a maximum of 350 Ci) to the torus once a day (the amount of tritium needed for each pulse, including gas puffing, is 44 Ci);
- keep tritium release level as low as is reasonably achievable;
- provide protection against potential accidental releases.

XI. NEUTRON ACTIVATION

A recent study [56] of the activation of the Ignitor experiment has been performed by modeling the machine with the neutron transport code ANISN [57] and evaluating the induced radioactivity with the inventory code ANITA [58]. The input data correspond to the reference parameters reported in Tables I and II. In particular, a 14 MeV neutron production of $\simeq 3.2\times10^{19}$ n/s (corresponding to a D–T discharge with $P_\alpha \simeq 18$ MW) and a neutron wall loading of $\simeq 2$ MW/m^2 have been assumed.
The results show that:

- the neutron–induced radioactivity in the Ignitor components has a major role in the maintenance and waste aspects of the machine;
- the short and long–term activation of the first wall is always negligible due to the benign characteristics of the graphite relative to activation;
- the short–term activation of the plasma chamber and magnet materials 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. In particular, after

40 sec of operation at the maximum power levels, the contact dose rate becomes unacceptable for direct handling;

- the long–term activation of the plasma chamber material (INCONEL 625) is such that its hands–on recycling could be prevented. In this case, however, the material can be classified as low level waste. All the other components present low long–term activation and can be recycled, if required;
- no practical solution is available to reduce the short–term activation of the inner components to levels permitting future immediate direct handling;
- radical changes in the adopted material for the plasma chamber would be required to reduce the long–term activation of this component.

XII. TRITIUM INVENTORY

The first wall material (graphite) and the plasma chamber material (INCONEL 625) have a quite different behavior regarding to tritium transport. For the first wall it is possible to carry out only an empirical analysis, while for the plasma chamber, made of a metallic material, analytical results will be given [59].

Graphite is essentially a porous material so that gaseous diffusion through pores and atomic diffusion on pore surface are intrinsically important to evaluate tritium retention in graphite. In addition to it porous structure, commercially available graphite usually has very wide variation in grain size and a complex grain structure which is not well characterized. There is a large scatter in the data on the interaction of tritium with graphite and no adeguate model has been developed yet.

The numerical solution of tritium transport in the plasma chamber have been obtained by the DIFFUSE code [60].

A. First Wall

Four distinct mechanism [61] have been identified for tritium retention in graphite:

- trapping in a saturated layer with a depth equal to the ion tenetration depth;
- diffusion along interconnected porosities;
- true lattice diffusion;
- co–deposition of tritiated carbon film, with subsequent adsorption and implantation of tritium into the deposited film.

At room temperature the saturated layer has a concentration of 0.4 T-atoms per C-atoms. This layer disappears at temperature near 1000 K as a result of the diffusion process. In Ignitor, the saturation layer will be a transient phenomena during a disruption event. For ions with energy between 200 to 300 eV, tritium tenetration depths are about 20 nm in graphite. Thus, the saturated surface region is able to retain about 10^{21} T/m^2 [62]. The surface regions will saturate rapidly because of the relatively low energy and high flux of tritium on the first wall. Glow discharge cleaning should be able to release the tritium trapped in the saturation region to a depth of about 5 nm [] (25% efficiency).

Experiments show that diffusion along the surface of interconnected porosity requires a lower activation energy (0.5 eV) than intragranular diffusion (2 eV). Therefore, diffusion is negligible at low material temperature (600 K) and retention reaches a maximum at 1400 K [,64] (about 5×10^{21} T/m^3 [62]). Moreover, taking into account the high activation energy for desorption (2 to 4 eV [62]), we may consider that tritium bulk retention is irreversible. In particular, bulk temperatures higher than 1700 K are required to release the trapped tritium. Therefore, there is no practical method to recove this tritium. On the other hand, the bulk trapped tritium does not present any risk of accidental release [63].

Co–deposition of tritiated carbon film takes place on all direct plasma–facing components (retention is about 5×10^{21} T/m^2 [62]), as well as on sheltered surfaces (the edges of the graphite tiles) with a retention estimated as 5×10^{22} T/m^2 [62]. Co–deposition is a reversible tritium retention mechanism. Cleaning methods such as plasma etching with exposure to CH_4/H_2, or He-5% O_2 glow discharges are expected to completely remove the co-deposited tritium [62,63]. The removal rate, as reported in [63], is only limited by the pumping speed of the torus. Therefore, it is merely a question of starting a cleaning phase whenever the tritium inventory reaches the allowed limit. In BPX [63] an overnight cleaning was expected to be able to remove the co–deposited layer from a day's operation.

A basic estimate of the maximum tritium inventory in the first wall is shown in Table IX, using the retention estimates reported above.

TABLE IX: Tritium Retention Estimates In The First Wall

	DATA
36 m^2	Plasma Facing Area
48 m^2	Sheltered Area
0.72 m^3	Graphite Volume
	ASSUMPTIONS
1×10^{21} T/m^2	Surface Saturation
5×10^{21} T/m^2	Co–deposition on Plasma-facing Area
5×10^{22} T/m^2	Co–deposition on Sheltered Area
5×10^{21} T/m^3	Bulk Retention
	REMOVAL EFFICIENCY
75%	for Surface Saturation
0%	for Bulk Retention
100%	for Co–deposition

	TRITIUM RETENTION (after cleaning)
0.14 gr. T	Surface Saturation
1.8 gr. T	Bulk Retention
0.0 gr. T	Co-deposition
2 gr. T	Maximum Tritium Retention

A. Plasma Chamber

The numerical results show that tritium inventory and permeation are negligible because of the low operating temperature and the absence of an incident flux of tritium. This result is important from a safety point of view, because it means that, in normal operation, there will be no release of tritium from this component.

XIII. MACHINE ASSEMBLY

The machine assembly [32] is to be done on site, using the following complete subcomponents:

- 12 moduli made up on 4 C-clamps; 2 TFCs; a sector of plasma chamber. The moduli are impregnated and bolted to accept a modicum of mishandling; the free surface of the TFC is sheathed by a layer of stainless steel to protect the insulation;
- the center post with the OHS coiled and impregnated on it. The handling may not be easy, but coiling the OHS on an extractable mandrel poses a greater risk of manipulation damage;
- the center post four heads in decomposed form;
- the shrink rings completely coiled and insulated;
- the pieces of the pretensioning system;
- the supporting legs;
- all PFCs, other than the OHS but including the press in a complete form;
- all structural glass fiber fittings and small pieces used, in many places, for insulating metallic components and/or transferring forces (a typical example is the thick cushion of glass fiber between the bracing ring tensioning devices and the C–clamps).

The first wall and the cryostat are arriving last and their assembly involves no external interfaces.

The assembly operation consists of building two halves, to be brought together and joined. The last two welds of the plasma chamber are to be performed from inside by the remote maintenance machine (this is the reason why there are entering lips in all welding seams). Then the PFCs and the shrink rings are installed, with the tensioning system. It is at this moment that the most important operation is to be performed to ensure that the three hyperstatic requirements of:

- wedging the TFCs inner legs;
- wedging the C–clamp;
- bucking the TFCs against the OHS

are simultaneously satisfied with the prescribed intensities. If the TFCs and the OHS were as rigid as the C–clamp wedged zone, the task would be impossible, but this is not the case since the insulated copper "gives" a lot. It is then necessary to check that the obtainable tolerances are much smaller than the elastic deformation obtained under ideal conditions; if this is the case (and indeed it is) then it is only a matter of following an appropriate assembly procedure, as described herewith. First, the rams are activated to produce some initial TFCs wedging: some shims between the TFCs contacting surfaces were purposefully added in the shop, where careful measurement took place prior to shipping on site. At this moment, since the outer sedging has not yet taken place, the pressure of the bracing rings is entirely taken by the inner wedging. Therefore, the operation is completely under control.

The hole of the OHS is now well–shaped and it can be accurately reamed to perfect roundness with a portable reamer. It is not necessary to assure any predetermined diameter but to measure it very precisely.

The center post and OHS, up to this moment, is not on site, but is on a lathe to be trimmed to a diameter consistent with that of the hole. It is then sent to the site and inserted after which the rams are given, gradually and uniformly, their full strength. Before the outer wedging is obtained, the bracing ring pressure goes, according to well–known elastic partition rules, to inner wedging and bucking. When the C–clamps touch, any movement stops and all bracing ring pressure goes to external wedging.

The axial compression of the inner TFC legs is then obtained by the bracing ring pull. This effect is spontaneous and does not need to be particularly adjusted, since the TFC, encased in the moduli, have been carefully assembled and shimmed in order to contact the C–clamps practically everywhere.

Once the machine is properly closed the plasma chamber can be stabilized and all the tile carriers can be assembled in their predetermined and carefully recorded position, all electrical connections made, and the cryostat erected. The latter is a conventional glass fiber and plastic structure with, however, one important feature. Around the ram stems the cryostat is double walled so that the pre-load of the bracing rings can be readjusted (as may be required by some shake down in the composite structural elements) without heating the entire structure, which would be a very time consuming operation. The rams themselves are not housed permanently to the machine to avoid freezing their rubber seals: they must be reinstalled prior to retrimming the bracing ring pull.

XIV. OPERATIONAL GUIDELINES

All the considered plasma equilibrium configurations follow these guidelines[32]:

- magnetic equilibrium are produced without exceeding the current limit in any PFC;
- the volt second requirements are consistent with those that the PFCs can supply;
- the stresses in the OHS are well controlled by the "bucking" effect of the TFCs system;
- bucking can be provided without unwedging the TFCs;
- the force excited by the electromagnetic press in addition to the C–clamp "nose bending", provide adequate compression in the TFC inner leg;
- a disruption taking place at any time during the discharge does not produce unbearable out of plane forces;
- at the end of the current pulse the thermal stresses, including creep critieria, are acceptable and the thermal expansion of the OHS does not unwedge the TFCs;
- the rate of plasma current drop, at the end of the pulse does not exceed the limit value (if it is it may require excessive voltage from the thyristor bridges);
- the C–clamp stresses are not excessive;
- interlocks are required to avoid excessive current in the TFCs and/or in the OHS when the current in the OHS and/or TFCs is not high enough;
- interlocks are required in connection with the press activation.

XV. CONCLUSIONS

The set of features that make a high field, tight aspect ratio toroidal plasma confinement configuration suitable for a deuterium–tritium 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 induced 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 able to satisfy the physics requirements has been shown to be possible 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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List of Symbols

n_o	peak plasma density
n_{eo}	peak electron density
$n_{\alpha o}$	peak α-particle density
$\langle n_e \rangle$	volume averaged plasma density
T_{eo}	peak electron temperature
$\langle T_e \rangle$	volume averaged electron temperature
τ_E	energy replacement time
$n_o \tau_E$	confinement parameter
R_o	major radius of the plasma column
$a \times b$	minor radii of the plasma cross section
δ_G	triangularity of the plasma cross section
κ	elongation of the plasma cross section
I_p	plasma current in the toroidal direction
$\langle J \rangle$	volume averaged toroidal current density
I_θ	plasma current in the poloidal direction
(I_{BS})	bootstrap current
V_ϕ	loop voltage
B_T	vacuum toroidal field at R_o
ΔB_T	paramagnetic (additional) field produced by I_θ
$\bar{B}_p$	mean poloidal field
$I_p \bar{B}_p$	confinement strength parameter
B_{vert}	vertical field at R_o
V_o	plasma volume
S_o	plasma surface area
$\langle p \rangle$	volume averaged plasma pressure
Z_{eff}	plasma effective charge
m^o	toroidal mode number
ℓ_i	plasma internal inductance
β_p	poloidal beta
β	toroidal beta
q_o	central magnetic safety factor
q_ψ	edge magnetic safety factor
$Vol_{q=1}$	volume inside the $q = 1$ surface
W	internal energy
P_{OH}	ohmic power
P_α	α-power
P_J	injected heating power

P_B	bremsstrahlung radiation power
P_{IC}	cyclotron and impurity radiation power
P_L	total power losses
V-s	poloidal magnetic flux variation
χ_i	ion thermal diffusion coefficient
χ_i^{neo}	neoclassical ion diffusion coefficient
χ_e	electron thermal diffusion coefficient
χ_e^{nOH}	non ohmic component of χ^e
χ_e^{OH}	ohmic component of χ^e
β_p^{crit}	threshold value for beta poloidal
D	deuterium
T	tritium
ICRF	ion cyclotron resonance frequency
TFC	toroidal field coil
PFC	poloidal field coil
OHS	central ohmic solenoid
t_r	ramp up time
t_{ar}	time after current ramp
t_I	ignition time
t_d	ramp down time
Q	fusion performance parameter
τ_v	plasma vertical displacement growth time
θ	poloidal angle
ϵ	passive plate efficiency in slowing the vertical plasma motion
n_v	equilibrium field index
$\bar{T}_{surf}$	average first wall temperature
T_{surf}	surface first wall temperature
Γ_{fw}	thermal wall loading on the first wall
S_{fw}	total area of the first wall
f_{rad}	fraction of P_L due to radiative processes
SOL	scrape off layer
f_p	peaking factor for the convection and conduction in the SOL
f_{con}	fraction of P_L due to convection and conduction in the SOL
f_{eff}	geometrical peaking factor (effective area of contact)
TSDS	tritium storage and delivery system

REFERENCES

1 B. Coppi, "High Current Density Tritium Burner," Report RLE PRR-75/18, Massachusetts Institute of Technology, Cambridge, MA, (1975) and *Comm. Plasma Phys. Cont. Fusion*, **3**, 2 (1977).

2 B. Coppi, and F. Pegoraro, *Il Nuovo Cimento*, **9 D**, 691 (1987).

3 B. Coppi, R. Englade, M. Nassi, L.E. Sugiyama, and F. Pegoraro, *Proc. 13th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research*, Washington D.C., USA, 1990, I.A.E.A., Vienna, **2**, 337 (1991).

4 B. Coppi, M. Nassi, and L.E. Sugiyama, *Physica Scripta*, **45**, 112 (1992).

5 L.E. Sugiyama, and M. Nassi, *Nucl. Fusion*, **32**, 387 (1992).

6 B. Coppi, M. Nassi, and L.E. Sugiyama, "Physics Principles of Ignition Experiments," invited paper at the Int. Sherwood Theory Meeting, Santa Fe, NM (1992).

7 B. Coppi, L.E. Sugiyama and M. Nassi, *Fusion Technology*, **21**, 1612 (1992).

8 B. Coppi, L. Lanzavecchia,*Comm. Plasma Phys. Controll. Fus.*, **11**, 47 (1987).

9 B. Coppi, *Vuoto*, **18**, 153 (1988).

10 B. Coppi, and The Ignitor Group, *Proc. 12th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research*, Nice, France, 1988, I.A.E.A., Vienna, **3**, 357 (1989).

11 ENEA, "Ignitor Project Feasibility Study," Rome, Italy (1989).

12 A. Angelini, B. Coppi, M. Nassi, *Proc. 14th IEEE/NPSS Symp. on Fus. Eng.*, San Diego, CA (1991), **1**, 411 (1992).

13 B. Coppi, and M. Nassi, *Fusion Technology* **21**, 1607 (1992).

14 G. J. Boxman, B. Coppi, L. C. De Kock, et al., *Proc. 7th Eur. Conf. on Plasma Physics*, 1975, Ecole Polytechnique Fédérale de Lausanne, Switzerland, **2**, 14 (1976).

15 A. C. Coppi and B. Coppi, *Nucl. Fusion*, **32**, 205 (1992).

16 B. Coppi, A. Taroni, G. Cenacchi, in Plasma Physics and Controlled Nuclear Fusion Research 1976, *Proc. of the 6th Int. Conf. Berchtesgaden*, **1**, 487 (1977).

17 S. C. Jardin, N. Pomphrey and J. Delucia, *J. Comp. Phys.*, **66**, 481 (1986).

18 M.E. Mauel, G.A. Navratil, S.A. Sabbagh et al., "Achieving High Fusion Reactivity in High Poloidal Beta Discharges in TFTR", *Proc. 14th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research*, Wurzburg, Germany, 1992, paper I.A.E.A. -CN-56/$A_I II - 4$.

19 P. J. Lomas, B. J. Green, J. How and THE JET TEAM, *Plasma Phys. Cont. Fusion* , **31**, 1481 (1989).

20 P.J. Lomas, B.J. Green, J. How, and JET Team, *Bull. Am. Phys. Soc.*, **32**, 1837 (1987).

21 D.J.Campbell, E. Lazzaro, M.F.F. Nave, et al., *Nucl. Fusion*, **28** , 981 (1988).

22 D. Campbell, J. G. Cordey, A. W. Edwards et al., *Proc. 12th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research*, Nice, France, 1988, I.A.E.A., Vienna, **1**, 377 (1989).

23 B. Coppi, P. Detragiache, S. Migliuolo et al., *Phys. Rev. Lett.*, , 2733 (1989).

24 J. Manickam, D. Roberts, R. Kaita et al., *Proc. of the 1990 Int. Sherwood Theory Conf.*, (Publisher College William and Mary), Williamsburg, VA (1990).

25 M. N. Bussac, R. Pellat, D. Edery et al., *Phys. Rev. Lett.*, **35**, 18 (1975).

26 H. Ara, B. Basu, B. Coppi et al., *Ann. Phys.*, **112**, 443 (1978).

27 M. Kaufman, K. Behringer, G. Fussman et al., *Proc. 12th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research*, Nice, France, 1988, I.A.E.A., Vienna, **1**, 229 (1989).

28 U. Stroth, G. Fubmann, K. Kreiger et al., *Nucl. Fusion*, **31**, 2291 (1991).

29 R. Betti and J. R. Freidberg, *Phys. of Fluids B*, **4**, 1465 (1992).

30 W. W. Heidbrink, E. J. Strait, E. Doyle et al., *Nucl. Fusion*, **31**, 15 (1991).

31 M. Nassi, "Poloidal Flux Requirement: Analysis and Application to the Ignitor Configuration," submitted to *Fusion Technology.*

32 C.I. Ferrari, "The Engineering Characteristics of the Ignitor Fusion Experiment: a Motivated Description", Consorzio CITIF, Turin, Italy (1990).

33 Consorzio CITIF, Report n. 1, Contract 150005, Turin, Italy (1992).

34 Consorzio CITIF, Report n. 12, Contract 150005, Turin, Italy (1992).

35 G. Cenacchi, B. Coppi, L. Lanzavecchia and M. Rulli, "Ideal MHD Equilibria and Poloidal Field Magnet System for Compact Ignition Experiments," Report RLE PTP-88/14, Massachusetts Institute of Technology, Cambridge, MA (1988).

36 N. Suzuki, A. Aikawa, K. Hoshino et al., *Proc. 12th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research*, Nice, France, 1988, I.A.E.A., Vienna, **1**, 207 (1989).

37 M. Nassi, S.C. Jardin, C.E. Kessel at al., "Vertical Stability Analysis for the Ignitor Configuration," *Proc. 17th Symp. on Fus. Tech.*, Rome, Italy (1992), paper Z:33.

38 C.E. Kessel and S.C. Jardin, *Proc. 14th IEEE/NPSS Symp. on Fus. Eng.*, San Diego, CA (1991), **2**, 1027 (1992).

39 Consorzio CITIF, Report n. 10, Contract 150005, Turin, Italy (1992).

40 Consorzio CITIF, Report n. 13, Contract 150005, Turin, Italy (1992).

41 Consorzio CITIF, Report n. 2, Contract 150005, Turin, Italy (1992).

42 A. Angelini, B. Coppi, A. Koch, L. Lanzavecchia, and G. Volta, *Proc. 4th Fus. Reactor Design and Tech.*, Yalta, **1**, 231 (1987).

43 A. Angelini, H. Quack, U.S. Patent Nr. 4, 884, 409 Dec. 5, 1989.

44 Consorzio CITIF, Report n. 15, Contract 150005, Turin, Italy (1992).

45 Consorzio CITIF, Report n. 7, Contract 150005, Turin, Italy (1992).

46 Consorzio CITIF, Report n. 8, Contract 150005, Turin, Italy (1992).
47 Consorzio CITIF, Report n. 5, Contract 150005, Turin, Italy (1992).
48 Consorzio CITIF, Report n. 6, Contract 150005, Turin, Italy (1992).
49 Consorzio CITIF, Report CTFFIGNN5251, Turin, Italy (1991).
50 S. Dinkevich, J. Swanson, T. Feng, et al., *Proc. 14th IEEE/NPSS Symp. on Fus. Eng.*, San Diego, CA (1991), **2**, 980 (1992).
51 Consorzio CITIF, Report n. 4, Contract 150005, Turin, Italy (1992).
52 C. Ferro and R. Zanino, "Edge Parameters and Scrape-off Layer (SOL) Characteristics in Ignitor," ENEA Report RT/NUCL/90/31, ENEA CRE Frascati, Italy (1990),*Proc. 9th Int. Conf. on Plasma Surface Interaction*, Bournemouth, U.K. (1990).
53 R. Zanino and C. Ferro, ENEA Report RT/FUS/89/26, ENEA CRE Frascati, Italy (1989).
54 J. L. Terry, E. S. Marmar, R. B. Howell et al., *Proc. 13th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research*, Washington D.C., USA, 1990, I.A.E. A., Vienna, **1**, 393 (1991).
55 A. Conte, W.T. Shmayda, N.P. Kherani, G. Bonizzoni, "Ignitor Tritium Storage and Delivery System", IFP-CNR Report FP 91/16, Milan, Italy (1991).
56 M. Zucchetti, and B. Coppi, *Fusion Technology*, **21**, 2017 (1992).
57 W.W. Engle jr., "A User's Manual for ANISN", ORNL K-1693, Oak Ridge Nat. Lab. (1967).
58 C. Ponti, S. Stramacchia, "ANITA Analysis of Neutron Induced Transmutation and Activation", EUR-12622 EN, Eur. Atom. En. Commun. (1989).
59 M. Nassi, "Tritium Inventoyr Estimates for the First Wall and the Vacuum Vessel of the Tokamak Ignitor", IFP-CNR Report FP 88/13, Milan, Italy (1988).
60 M.I. Baskes, SAND Report 83-823, (1983).
61 K. Ichimura, T. Tanabe, K. Wilson, M. Yamawaki, *Proc. Int. Conf. Plasma Wall Inter.*, Nagoja, Japan (1987).
62 H.F. Dylla, " First Wall Conditioning and Tritium Inventory", CIT Conceptual Design Review, (1988).
63 M. Ulrickson, J.N. Brooks, H.F. Dylla et al., *Fusion Technology*, **21**, 1279 (1992).
64 R.A. Causey, M.I. Baskes, K.L. Wilson, *J. Vac. Sci. Techn.*, **A5(4)**, 2768 (1987).