Critical Physics Issues for Ignition Experiments: Ignitor

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Abstract

The crucial physics issues related to fusion burning plasmas and potential fusion reactors can only be studied in a dedicated plasma experiment that is designed to attain ignition conditions. The Ignitor experiment takes a conservative approach to the near term study of the physics of igniting plasmas, using an optimal combination of compact dimensions and high magnetic fields to support high plasma particle densities and high plasma currents. The values of its geometrical parameters, plasma current, and magnetic field have been chosen based on current knowledge of fusion burning physics, so that ignition can be achieved. The present paper describes the most important ideas motivating this experiment and their relation to the specific choices made in establishing the Ignitor project.

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1 Introduction

Demonstration of fusion ignition is a major scientific and technical goal for contemporary physics. Until the fundamental physics issues of fusion burning have been identified and confirmed by experiment, the defining concepts for a fusion reactor will remain uncertain. An important value of a basic ignition experiment is that the ignition process will be similar for any magnetically confined, predominantly thermal plasma. In such an experiment, heating methods and control strategies for ignition, burning, and shutdown can also be established.

These three issues, demonstration of ignition in a magnetically confined plasma, the physics of the ignition process, and heating and control of a burning plasma, are specifically addressed by the Ignitor experiment [1,2,3,4,5]. Its design has been driven primarily by physics considerations since its inception. The associated physics studies have gone beyond simple identification to include the interaction of the physical processes involved in ignition. Ignitor is part of a line of research that began with the Alcator machine at MIT in the 1970's [6,7], which pioneered the high magnetic field approach to plasma magnetic confinement, and continued with the Alcator C/C-Mod and the FT/FTU series of experiments. The idea for the first D-T ignition experiment proposed on the basis of existing technologies and knowledge of plasma physics was formulated at about the same time as the first results of the Alcator experiment were produced [8]. Subsequent developments have confirmed the fact that high magnetic fields combined with an optimized compact geometry offer, at present, the only path to achieving ignition, when both plasma energetics and stability are taken into consideration. The approach involving the combination of appropriate geometry and high magnetic fields also allows a possible development path [9,10] to tritium-poor, reduced-neutron-production fusion, which could yield interesting kinds of fusion reactors.

A considerable amount of work on the physics of ignition has been carried out over the course of the Ignitor design. Much of it is generally applicable to ignition in a confined plasma, not only at high fields. This article presents the basic physics that underlies the Ignitor design, including the open questions. It starts with the physics questions that cannot be addressed in present experiments, then discusses the problem of attaining ignition and the reasons for the necessity of high magnetic fields and compact configurations at this time. It then proceeds with the Ignitor parameters and reference operating scenarios, the dynamic nature of the ignition process and its relation to the initial current rise phase of a discharge, and other issues.

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2 Beyond present experiments

Even without strong assumptions on the possible characteristics of a fusion reactor, it is clear that the most advanced among present experiments do not operate in plasma regimes close to those required for ignition. There are a number of shortcomings, of which one or more always apply:

1. The plasma contamination as measured by the effective charge Z_{eff} is in general too high, compared to the limiting value for stable ignition, $Z_{eff} \lesssim 1.5$-1.6, that has been identified since the early analyses of the approach to ignition by a magnetically confined plasma [1,2]. In particular, relatively high Z_{eff} leads to excessive radiation emission, and increased plasma pressure is required by the higher nuclei dilution relative to the electron density. Although initially demonstrated for Ignitor, this Z_{eff} limit can be shown to be general. By exceeding this value, even before other prohibitive limits are reached, larger amounts of auxiliary heating power are required, operation near the β stability limit, and other obstacles begin to surface.
2. The central ion temperature is substantially higher than the electron temperature, $T_i > T_e$. A D-T thermal burning plasma in the temperature range that is commonly considered will have $T_i \leq T_e$, as fusion alpha particles and all other charged particles produced by fusion reactions have relatively high energies, in the MeV or multi-MeV range, and therefore primarily heat the electrons by collisional slowing down. In present experiments, the ions used for neutral beam heating have relatively low energy, on the order of 100 keV, and primarily heat the ions. In addition, they sustain a large fast ion population, due to the relatively long collisional slowing down times at the low plasma densities into which they are injected.
3. The alpha particle slowing down time is long compared to the energy confinement time τ_E, while in an igniting plasma it should be much shorter. The degree of single orbit confinement of the alpha particles is considerably smaller than that

characteristic of ignition experiments designed for plasma currents of 10 MA or higher.

4. Present experiments have relatively low peak plasma pressure. The ignition figure of merit $n_{i0}T_{i0}\tau_E$ requires in practice peak values of the plasma pressure $1 \lesssim p_0 \lesssim 4$ MPa. For D-T fusion, $n_{i0}T_{i0}\tau_E \simeq 60$ (in units of $10^{20}\,\mathrm{m}^{-3}$, keV, sec). More accurate figures of merit, for example $n_{H0}T_{e0}\tau_E\varepsilon_{sd}\varepsilon_p$, could take into account the slowing down time of the fusion alpha particles, through ε_{sd}, which may be assumed to be of the form $\tau_E/(\tau_E+\tau_{sd})$, and the plasma purity through ε_p, which may be chosen as $5/(4+Z_{eff}^2)$.

5. The record power producing D-T experiments performed so far, in spite of their relatively low densities, have been ballooning unstable. High fusion yield discharges have consistently been quenched by plasma instabilities. In particular, in TFTR $n=1$ kink-driven edge ballooning modes have been identified as the instability responsible for the termination of the record discharges.

6. Most of the known improved confinement regimes are transient and/or nonthermal (significant non-Maxwellian particle distributions). Improved confinement regimes tend to be associated with modified, transient q-profiles, while most high confinement experiments using NBI heating have a substantial nonthermal ion population due to the relatively low bulk-plasma densities. A discussion of the regimes that do not fit this characterization is given in Section 8.

7. Referring to the onset of m = 1, n = 1 internal modes, the electron collision frequency ν_e is smaller than the "drift" frequency

$$\omega_{*e} \equiv -(1/r_0)(cT_e/eB)(dn/dr)/n$$

where $r=r_0$ indicates the surface with q = 1. In fact, the value of the ratio ν_e/ω_{*e} is important for the characteristics of these modes when the plasma poloidal beta β_p is still below the ideal MHD instability threshold. The discharges with $\nu_e/\omega_{*e}<1$ involve relatively low densities compared to an ignition experiment, where higher density is desirable to increase the fusion reaction rate and improve plasma purity. The collisionless reconnecting modes that can be excited for $\nu_e < \omega_{*e}$ are definitely milder than those expected in semi-collisional regimes, with $\nu_e \gtrsim \omega_{*e}$. Since D-T ignition experiments must operate in the latter regimes, the known stability criteria against collisional modes, that can reach significant amplitudes, have to be satisfied.

The plasma regimes in which Ignitor is planned to operate avoid all of the shortcomings that we have listed.

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3 The ignition objective

The first of the intermediate goals of an ignition experiment is to reach the ideal ignition temperature at which fusion heating begins to dominate the bremsstrahlung radiation losses (this occurs at peak temperature $T_{e0} \simeq T_{i0} \simeq 6\,\text{keV}$ for typical centrally peaked profiles), under conditions in which the fusion heating can continue to rise. To study actual ignition and true fusion burning, experiments must proceed further and operate in regimes with high levels of fusion power relative to other inputs, for times long compared to the most important plasma characteristic time scales. Writing the plasma power balance equation as $dW/dt = P_{\alpha H} + P_{Aux} + P_{OH} - P_L$, where W is the plasma thermal energy and P_{Aux} and P_{OH} the externally applied and ohmic heating powers respectively, we argue that reaching a condition where $Q_{\alpha H} \equiv P_{\alpha H}/(P_L - P_\alpha) \gtrsim 2$ is necessary in order to identify the main collective processes contributing to the energy balance. Here P_L represents the total power loss from the plasma and $P_{\alpha H}$ the fraction of the produced alpha particle power that actually heats the plasma. The definition of ignition used throughout the Ignitor work is $P_\alpha = P_L$, where P_α is the total alpha particle power $P_\alpha = P_F / 5 \geq P_{\alpha H}$, and P_F is the total power produced by fusion reactions. In the case of Ignitor this condition corresponds to an over-heated state with $dW/dt > 0$, as $P_{OH} \neq 0$. Thus, the temperature will first make an upward excursion before settling at a lower, appropriate level.

Ignitor uses high magnetic fields and high currents in a compact geometry, which allows ignition at relatively low fusion power levels and low plasma betas, with strong ohmic heating. These are not the only advantages of a high field approach and a strong argument can be made that the combination of high magnetic fields with optimized compact confinement configurations represents the only real solution for the ignition dilemma (Section 4) at the present time. This combination introduces an interlocking set of requirements [1] , which are summarized in Appendix A. The maximum values of the field and of the plasma current that can be generated and the length of time over which they can be sustained in a given magnetic configuration are thus strong factors to assess the capacity of the proposed experimental facility.

For reference, the basic parameters of the Ignitor device are given in Table 3.1, while a cross section of the machine is presented in Fig. 3.1. The pulse length can vary considerably depending on the fields and currents that are to be produced (Fig. 3.2). It takes approximately 4 sec to reach the maximum value of the plasma current without additional RF heating. The following flat-top phase is about 4 sec at 13 T and 11 MA (reference value) (Fig. 3.2a). In Fig. 3.4 we give a representation of the typical spatial dependence of the poloidal field in a 11 MA plasma. Examples of equilibrium configurations at 13 T and 11 MA are shown in Fig. 3.3a,b. At 12 MA, the current is slowly decreased to 10 MA after 1 sec to prevent a full current penetration and the consequent appearance of a considerable central region where $q < 1$ (Fig. 3.2b). At 9-10 T, the flat-top duration can last over 10 sec (Fig. 3.2c). Therefore, given the high densities that Ignitor can sustain, the machine is designed to operate over time scales considerably longer than the important intrinsic plasma time scales.

An extensive list of plasma parameters resulting from one of the numerical simulations by the JETTO code [2], in the version described in Ref. [3] for an ignition scenario at 11 MA is presented in Table 3.2. In this case, the efficiency for the alpha particle heating is assumed to be 1, in view of the high values of the plasma current, and the wall reflectivity for the syncrotron emission is taken equal to 0.9. The numerical coefficient for the CMG electron thermal diffusivity is close to unity, the ion conductivity is neo-classical with an additional term equal to a fraction (5%) of the electron diffusivity, to obtain realistic profiles. We note that the CMG coefficient gives realistic temperature profiles but does not represent the best confinement conditions achieved in high density experiments with ohmic heating only. The approach to ignition conditions is shown in Fig. 3.5.

The advantage of low-temperature ignition (we take 11 keV as a reference value) is self evident from Fig. 3.6: should the confinement time be less than expected, ignition can still be reached at slightly higher temperatures. Specifically, the figure gives $\tau = W / P_\alpha$ as a function of the peak temperature T_0, obtained for a class of easily represented density and temperature profiles that are relevant to Ignitor. Here W is the total plasma thermal energy evaluated with $T_i = T_e$ as 3/2 of the volume integral of $p = (p_i + p_e)$. The fusion power P_α is estimated as $\varepsilon_\alpha \int dVn^2 \langle \sigma v \rangle_{fus} / 4$. The radial profiles are represented by generalized parabolas with an exponent, γ_T and γ_n for the temperature and density respectively. The value n_0 of the peak density is kept fixed at 10^{21} m^{-3}. The small width of the curve indicates that significant variations of the peaking parameters γ_T and γ_n have a rather modest effect. The arrows indicate the reference temperature of 11 keV and the one, about 14.5 keV, that corresponds to a confinement time degraded by 2/3 relative to that at 11 keV. This is to show that Ignitor can fulfill its objectives even if the energy confinement time is seriously degraded relative to current expectations, before being limited by stability problems associated with the value of the peak pressure.

As is evident from the machine layout and from Fig. 3.3, a highly flexible poloidal filed coil system has been adopted, such that it has the ability to produce a considerable variety of equilibrium configurations. Its original design was conceived as a development of the system adopted on the DIII-D machine, using the codes that reproduce accurately this machine actual equilibria and their relationship to the distribution of the currents in the various poloidal field coils. The variation of magnetic flux, linked with the plasma column, that the poloidal field system can produce is estimated around 33 volt-sec.

The injection of 18 – 24 MW of RF power at the ion cyclotron range of frequencies ($\simeq$ 70 – 140 MHz) is an integral part of the machine design. For this, 6 of the 12 large equatorial ports are utilized. The purpose of the injection heating system is to extend the region of parameter space where ignition can be achieved or other interesting plasma regimes can be accessed and to gain more control over the evolution of the temperature and current density profiles, relative to the case where only ohmic heating is used. An important factor associated with the application of ICRH is that the time needed to reach ignition can be shortened and therefore the time interval available to study the properties of ignited plasmas can be extended.

A high speed pellet injector, for the purpose of attaining the proper density values and profiles, is also an important component of the machine design.

The adopted magnets are supercooled, with temperatures starting at the optimal value, for copper, of 30 K (Fig. 3.7). Gas helium is employed for the cooling that takes place between pulses. Therefore the duty cycle depends on the maximum temperature reached by the magnets at the end of each pulse. We note also that the toroidal field magnet is split into 24 modules ("coils") in order to ensure a low field ripple at the outer edge of the plasma column [4].

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Table 3.1: Ignitor Reference Design Parameters

major radius	R_0	$1.32\,\mathrm{m}$
minor radius	$a \times b$	$0.47 \times 0.86\,\mathrm{m}$
aspect ratio	A	2.8
elongation	κ	1.83
triangularity	δ	0.4
toroidal field	B_T	$\lesssim 13\mathrm{T}$
toroidal current	I_p	$\lesssim 11-12\ \mathrm{MA}$
maximum poloidal field	$B_{p,max}$	$\lesssim\ 6.5\ \mathrm{T}$
mean poloidal field	$\overline{\overline{B}}_p \equiv I_p / 5\sqrt{ab}$	$\lesssim 3.44-3.75\ \mathrm{T}$
poloidal current	I_θ	$\lesssim 9\,\mathrm{MA}$
edge safety factor ($I_p \simeq 11\,\mathrm{MA}$)	q_ψ	3.6
confinement strength	$S_c \equiv I_p \overline{\overline{B}}_p$	38 – 45 MA·T
plasma volume	V_0	$\simeq 10\ \mathrm{m}^3$
plasma surface	S_0	$\simeq 34\ \mathrm{m}^2$
ICRF heating ($\simeq 140\,\mathrm{MHz}$)	P_{RF}	18 – 24 MW

Table 3.2 Example of Plasma Parameters at Ignition

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}
Plasma Stored Energy W	11.9 MJ
Ohmic Power P_{OH}	11.2 MW
ICRF Power P_{ICRH}	0
Alpha Power $P_{\alpha H}$	19.2 MW
Bremsstrahlung Power P_{brems}	3.9 MW
Poloidal Beta β_p	0.2
Toroidal Beta β_T	1.2 %
Central $q(\psi)$ q_0	~ 1.1
Edge q_ψ	3.5
Bootstrap Current I_{bs}	0.86 MA
Energy Confinement 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

$\langle\ \rangle$ = volume average

$$\beta_p = 2\mu_0 \langle p \rangle \Big/ \overline{B}_p^2(a)$$

$$\tau_E = W / (P_{OH} + P_\alpha + P_{ICRH} - dW/dt)$$

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

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

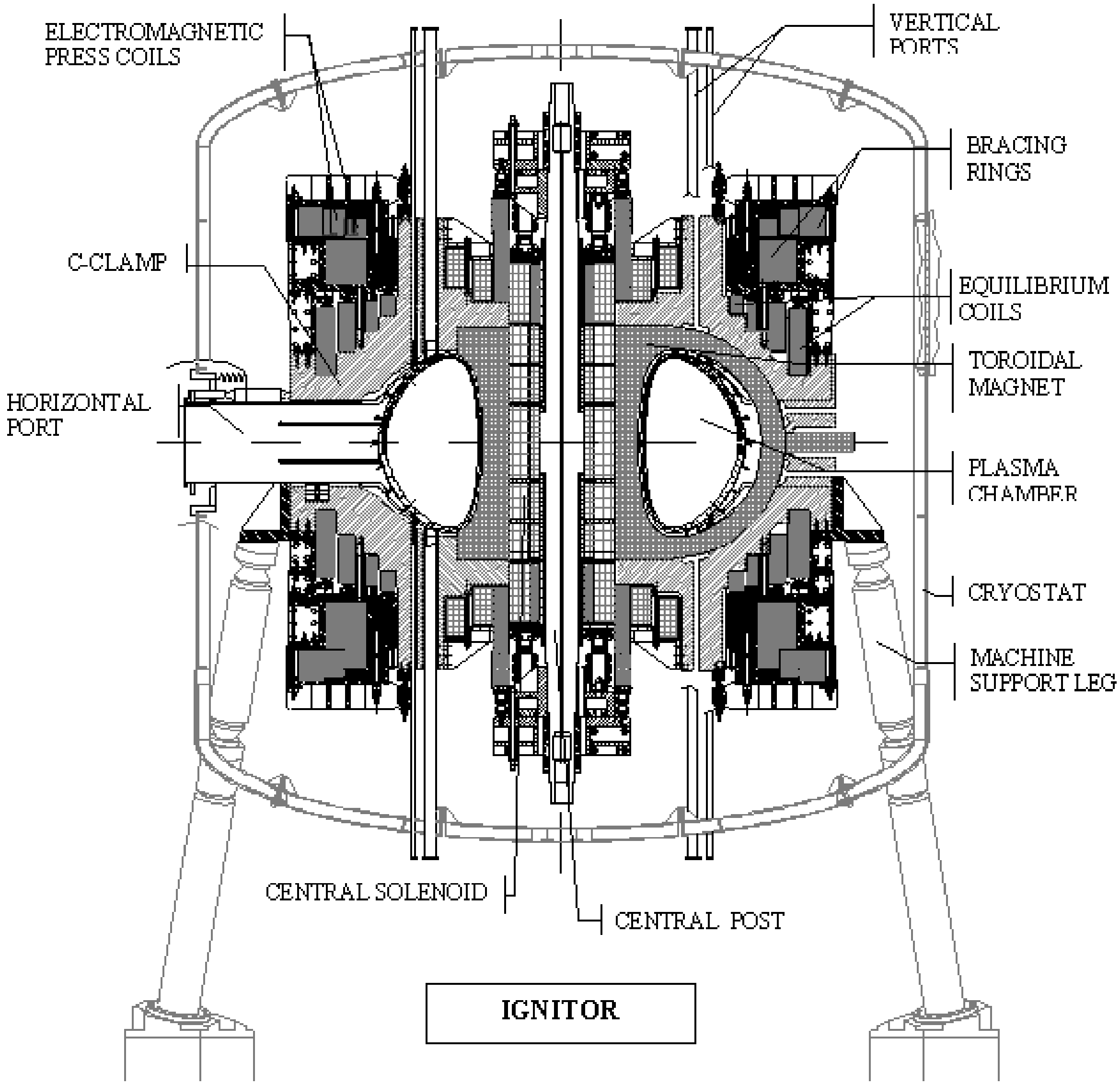
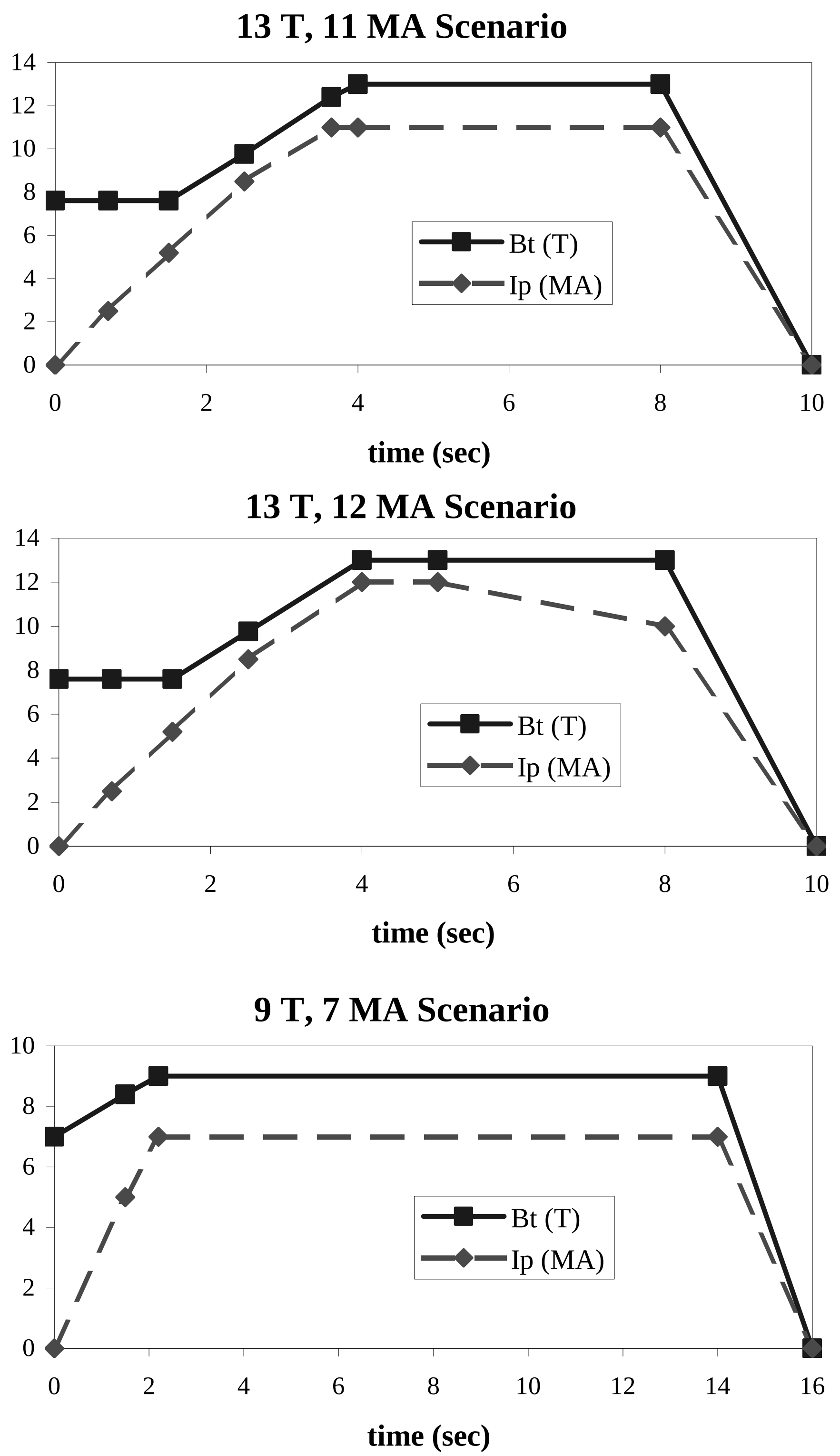


Fig. 3.1. Vertical cross section of the Ignitor machine.



a)

b)

c)

Fig. 3.2. Examples of operating scenarios.

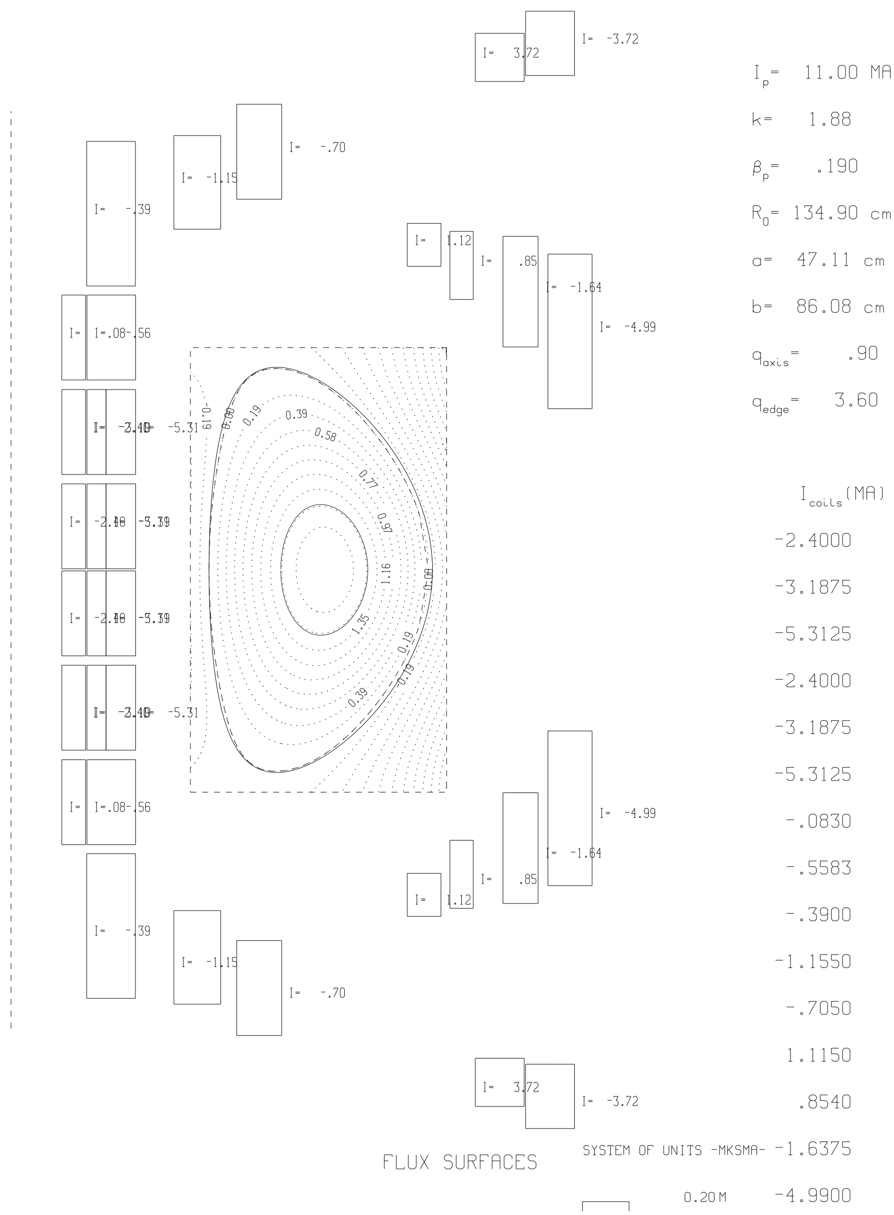


Fig. 3.3a. Equilibrium configuration for 11 MA and 13 T, as evaluated by the EQUISL code, with $q_0 \simeq 0.9$.

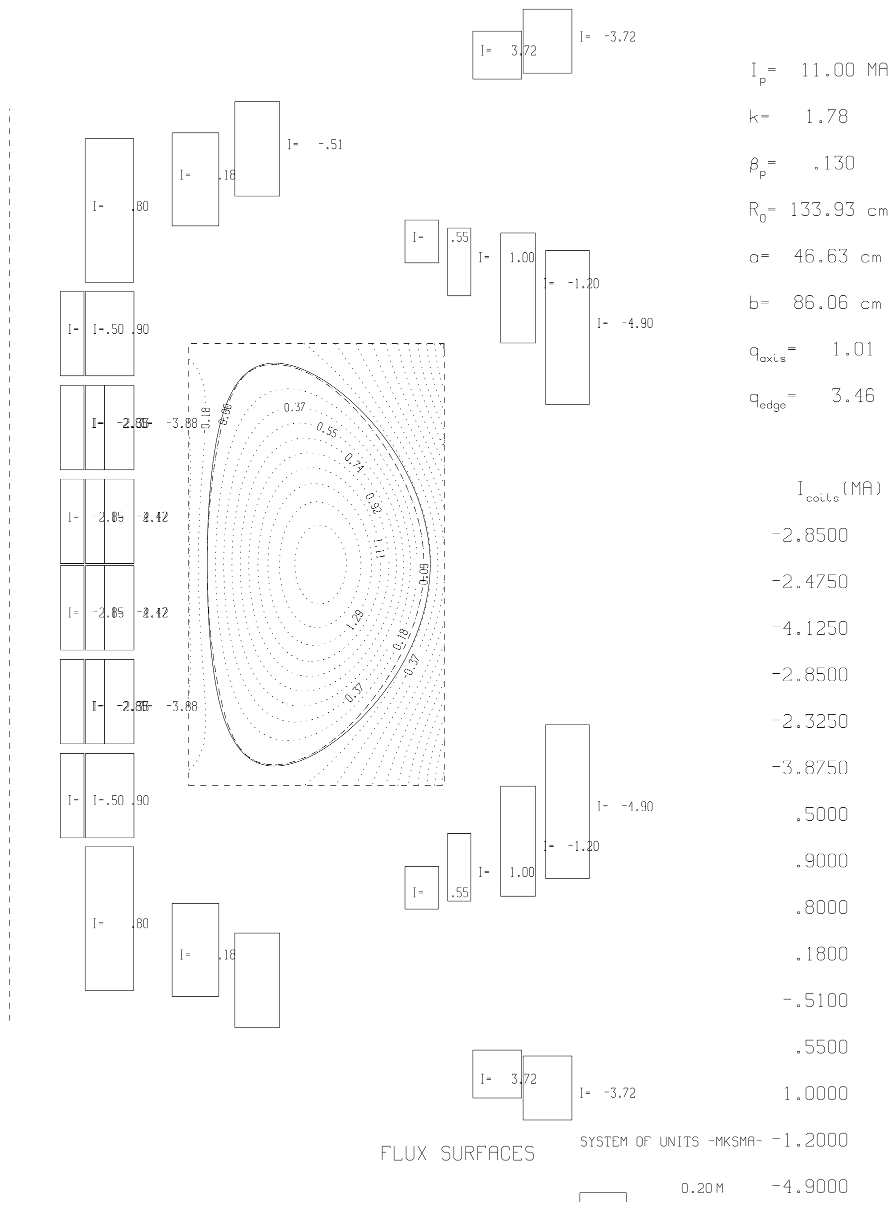


Fig. 3.3b. Equilibrium configuration for 11 MA and 13 T, as evaluated by the EQUISL code, with $q_0 \simeq 1$.

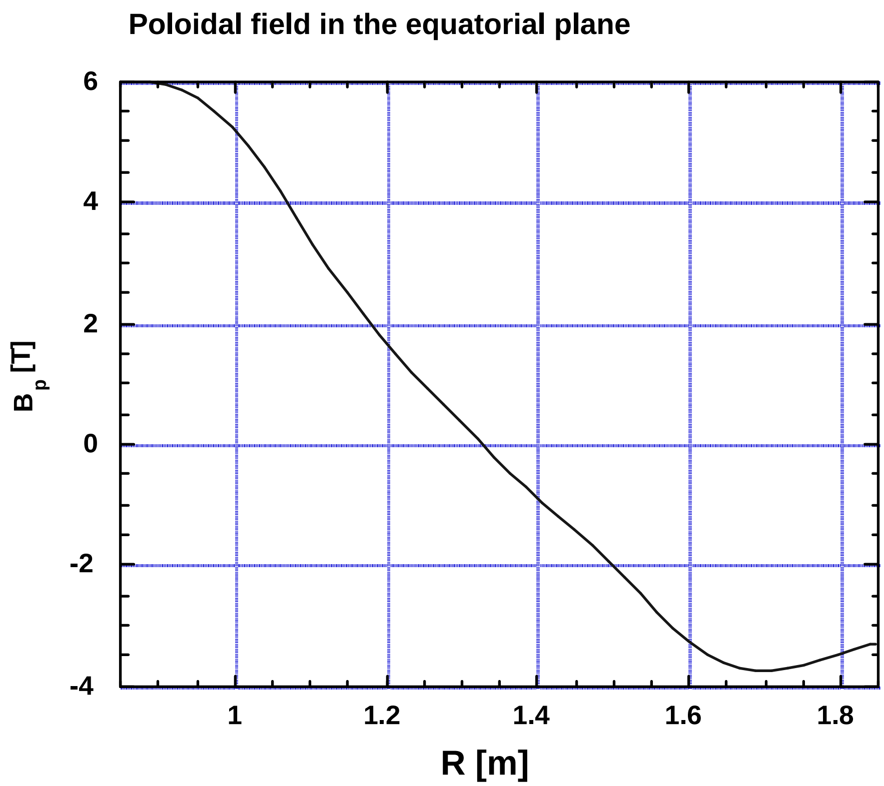
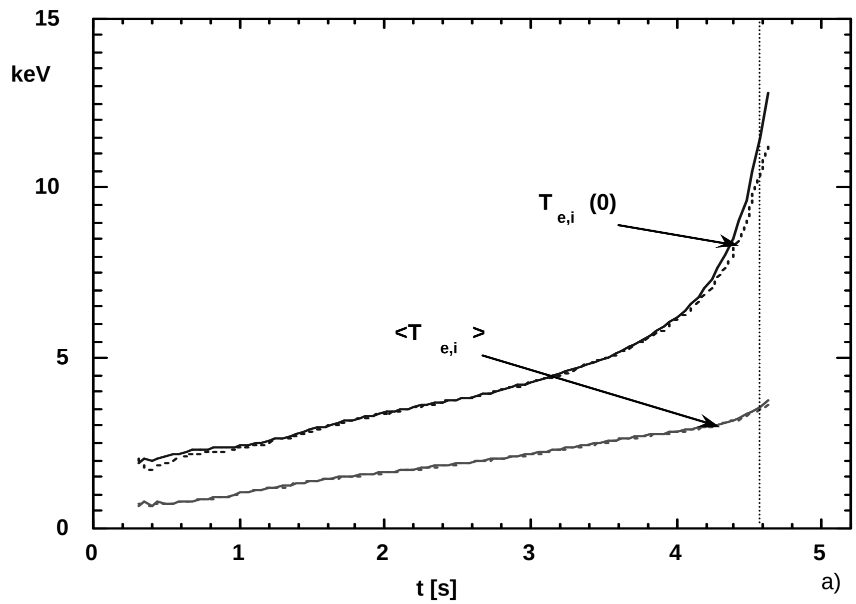
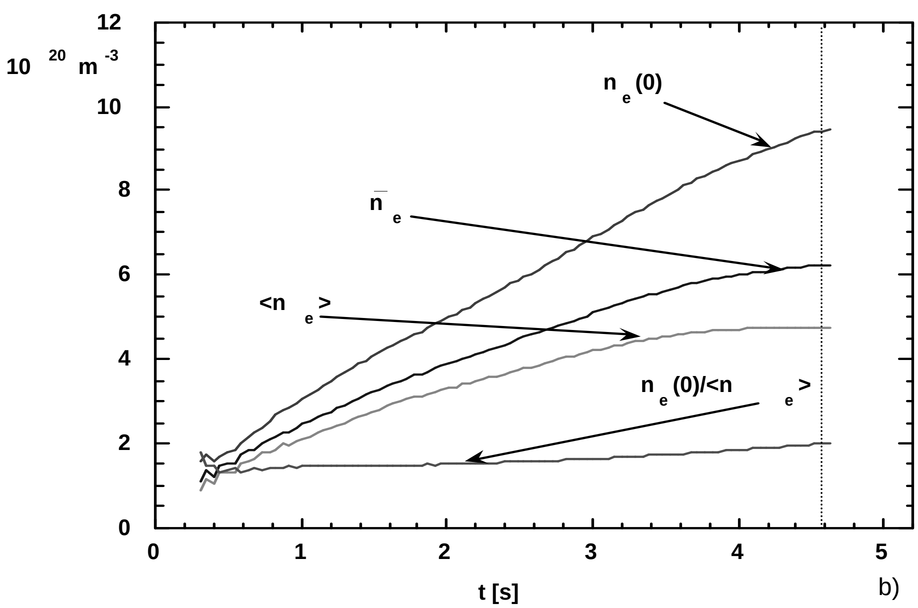


Fig 3.4. Dependence of the poloidal field B_p , for an 11 MA scenario, on the distance from the axis of symmetry.


15
keV
10
5
0
0
1
2
3
4
5
t [s]
$T_{e,i}$ (0)
<$T_{e,i}$>

a)


12
10^{20} m^{-3}
10
8
6
4
2
0
0
1
2
3
4
5
t [s]
n_e(0)
$\bar{n}_e$
<n_e>
n_e(0)/<n_e>

b)

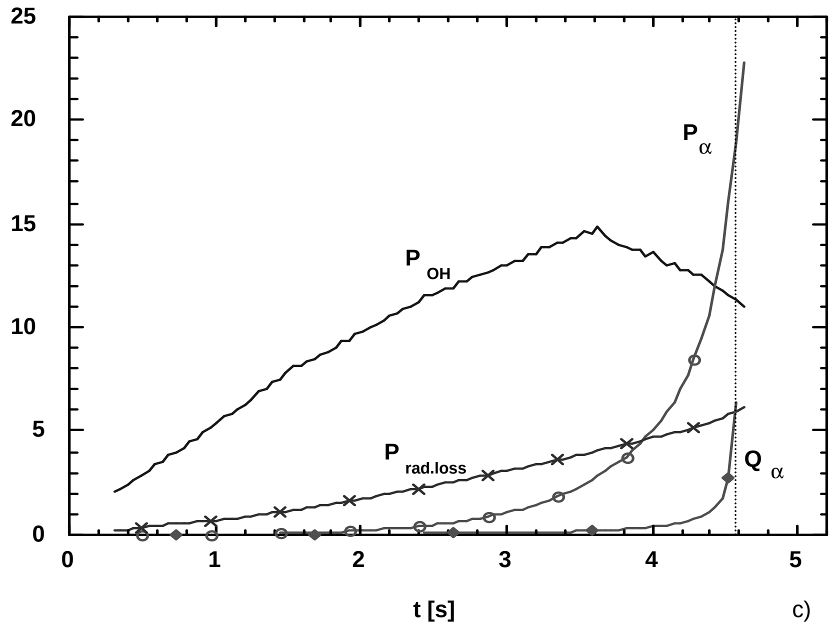


Fig 3.5. Time evolution of a discharge with 11 MA and 13 T simulated by the JETTO code, showing a) temperature, b) density, c) powers in MW and the parameter Q_α. Ignition is marked by the vertical dotted line.

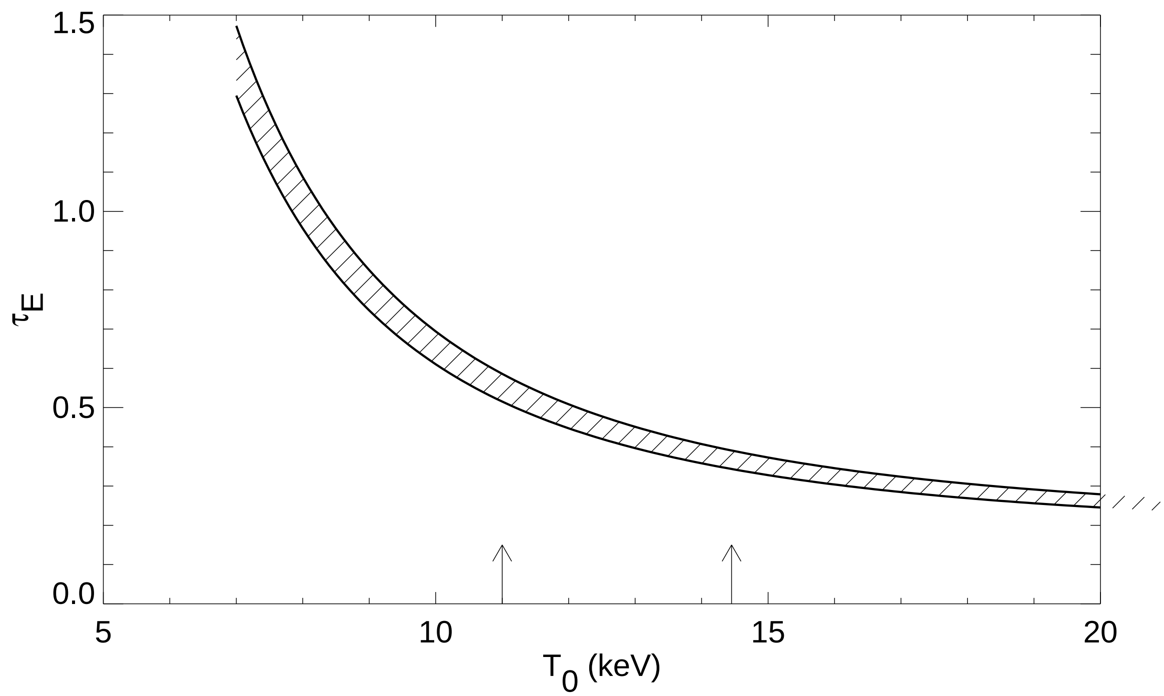


Fig. 3.6. Confinement time at ignition as a function of the peak temperature, for a fixed value of the peak density, $n_0 = 10^{21}\ m^{-3}$*, assuming equal ion and electron temperatures. The temperature profile peaking factor* γ_T *is varied from* 1 *to* 2.5*, while the density factor* γ_n *goes from* 0.5 *to* 2*. The reference ignition temperature of* 11 *keV corresponds to the average of the electron and the ion peak temperatures in Table* 3.2*. The corresponding energy replacement time is* $\tau_E \simeq 0.56$ *s. The second arrow corresponds to the peak temperature (*$\simeq$ 14.5 *keV) necessary to ignite when the energy confinement time is reduced by* 2/3 *(that is* $\tau_E \simeq 0.37$ *s). We note that this corresponds to a substantial increase of the produced fusion power.*

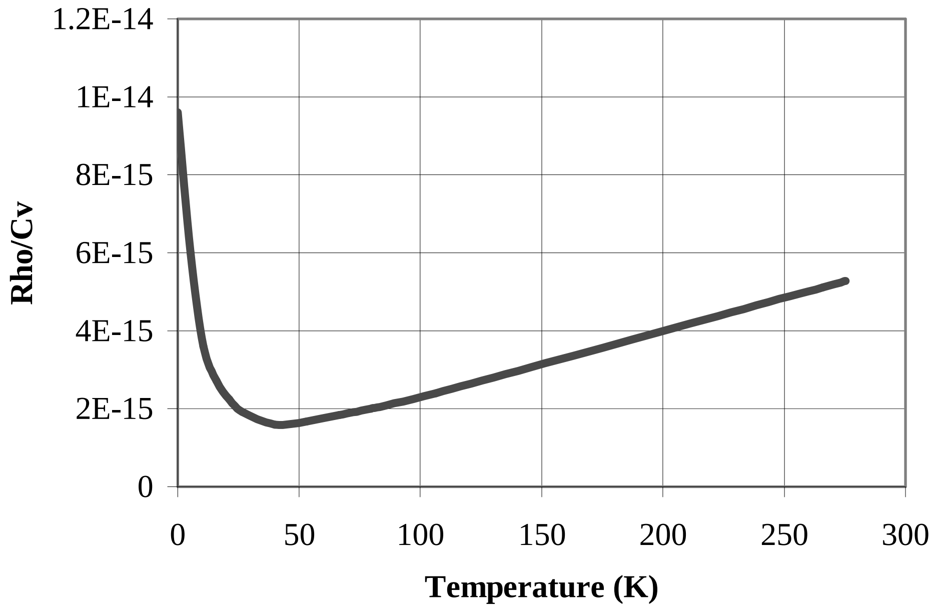


Fig. 3.7. Ratio of resistivity to specific heat at 20 T for the copper material adopted for the toroidal magnet.

4 Physics conditions to ensure ignition: high B_T and optimized compact configurations

The many physics limitations and uncertainties regarding ignition (see also the Snowmass Burning Plasma Report [1]) lead to the statement that "High magnetic field is the most advantageous approach to ignition using the present knowledge of the physics and technology of high temperature plasmas." This conclusion also emphasizes the importance of continuing technological progress, such as the development of superconductors capable of sustaining fields of 20 T or more in magnets of significant size. In fact, a combination of stability and transport considerations related to the confinement of plasmas that are capable of igniting points to the stronger conclusion "High magnetic field combined with optimized compact confinement configurations is the only possible approach to ignition at this time."

To justify this we consider the possible values of the edge safety factor q_ψ, the required central pressure and values for β_p that are consistent with both ignition conditions and macroscopic plasma stability. Stability considerations in practice require the poloidal plasma beta, $\beta_p \equiv 2\mu_0\langle p\rangle / \bar{B}_p^2$, where $\bar{B}_p$ is the flux surface averaged poloidal field, to be less than a critical value $\beta_{p,crit.}$ For an estimate of $\bar{B}_p$ we may use the following expression $\bar{B}_p = (a/R_0)\left(B_T/q_\psi\right)G_1$ where G_1 depends on the geometrical characteristics of the plasma column $(G_1 \approx 2.5)$. At ignition, the minimum central pressure p_0 has to be in the range of $1 \lesssim p_0 \lesssim 4$ MPa for 50:50 D-T plasmas (1 MPa $\simeq$ 10 atm) as indicated earlier.

There are two characteristic regimes to pursue ignition, with low and high q_ψ. At low edge-q_ψ, e.g. $q_\psi \simeq 3$, (an approximate value), the regions where $q < 1$ and $q < 2$ are both relatively large. Then large scale internal modes with $n = 1$ and dominant $m = 1$ and $m = 2$ harmonics, extending to r_1 and r_2 respectively, will exist unless β_p is also small. Since the volume average $\langle p\rangle$ cannot be too low at ignition, plasma stability requires a minimum B_T that depends on the critical $\beta_{p,crit}$. Thus we obtain:

$$\bar{B}_p > \left[\frac{2\mu_0\langle p\rangle}{\beta_{p,crit}}\right]^{1/2} \simeq \left(\frac{\langle p\rangle_{MPa}}{\beta_{p,crit}/0.3}\right)^{1/2} \times 2.9\,\mathrm{T}$$

and

$$B_T \gtrsim \sqrt{\frac{p_{0,MPa}/3}{\beta_{p,crit}/0.3}}\left(\frac{3\langle p\rangle}{p_0}\right)\left(\frac{R_0}{3a}\right)\left(\frac{q_\psi}{3}\right)\left(\frac{2.5}{G_1}\right)\times 10.4\,\mathrm{T}$$

At high edge-q_ψ, such as $q_\psi > 5$, the plasma current must be relatively low, as $I_p = 5a^2\sqrt{\kappa}B_T / \left(Rq_\psi G_2\right)$ where B_T is measured in tesla, I_p in megamperes and G_2 is an appropriate geometrical factor. Assuming that the confinement time is $\tau_E \propto I_p$, as is typical of most confinement scalings, then considerably high values of the confinement improvement factor *H* over L-mode are required to reach ignition. In practice, *H* is observed to be limited to values of 2-3. We may argue that for a given confinement configuration and fixed values of q_ψ, $\bar{B}_p \propto B_T$. Therefore, the factor $a\sqrt{\kappa}B_T$ cannot be too small. If $a\sqrt{\kappa}B_T$ is increased by expanding the radius, the average poloidal field $\bar{B}_p$ is relatively low if B_T is relatively low, and β_p tends to become relatively large. Pressure-gradient-driven ballooning modes then become a problem.

Applying the actual values from experiment and theory shows that these criteria give fairly stringent practical limits on B_T. In the end, values of $q_\psi \simeq 3.6$ such as those chosen for Ignitor correspond to the least restrictive conditions and are the best choice for ignition, given the limits on the achievable B_T with present day magnet technology.

References

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5 The transient nature of the approach to ignition

In a confined burning plasma, the approach to ignition is a transient process, where both spatial and temporal effects are important [1]. At the end of this process, as is well known, the strong positive dependence of the fusion cross sections on the kinetic energy of the reactants allows the possibility of a “thermonuclear instability” phase where the plasma temperature and fusion power can rise rapidly.

For magnetically confined plasmas, the transient nature of the approach to ignition is particularly important, because the initial growth phase of the plasma column when the plasma current is being raised to its final value (i.e., the current ramp phase) can be exploited to ohmically heat the plasma towards ignition at the same time. The development of the plasma profiles can be controlled, in particular the toroidal current density J_ϕ, in order to ensure plasma stability (see ref. [2] for initial current ramp studies, and [3,4,5] for integration of heating and plasma stability effects for Ignitor). An important constraint is the final value of the edge q_ψ allowed by the plasma field, current, and shape. A great deal of work for Ignitor has been done to confirm that this procedure can be effective and to study its limitations (e.g., [5,6,7,8]). Much of this work predated later successful experiments showing that improved confinement regimes could be obtained through control of the current ramp (the early Ignitor work did not consider such regimes and actually imposed the condition that the *q*-profile remain monotonically increasing toward the plasma edge; reversed shear and improved confinement was considered in [7]).

Understanding the transient approach to ignition is a complex problem, since a large number of independent or semi-independent time-varying parameters must be optimized. A numerical transport simulation model containing at least the radial (flux-surface) coordinate is required for quantitative results. The basic principles are clear, however.

In the case of Ignitor, transient effects are exploited so that ohmic heating can by itself alone carry the plasma to ignition or, in any case, give a substantial boost toward ignition [4,5]. In particular, during the current ramp phase the plasma current is increased by adding “skin layers” of current, to the outer surface of the plasma column, that are not given the time to diffuse inward. The plasma loop voltage then peaks at radii near the edge of the plasma, a region of relatively large volume (cf. the relevant figures in [5]). Consequently, since the collisional resistivity $\eta_\parallel$ is proportional to $T_e^{-3/2}$, a relatively large ohmic heating power $I_p V_\parallel$ can be produced even when the central plasma temperature is high. The ohmic heating rises continuously during the current ramp, with a power P_{OH} that is roughly proportional to I_p. For the Ignitor reference scenario, this power varies from about 15 MW near the end of the current ramp to approximately 10 MW at ignition when, roughly, $P_{\alpha H} \gtrsim 2P_{OH}$. We note that due to the high field and current, self-sustained burning states can be reached and maintained by the residual ohmic heating at reduced levels of confinement, even if full ignition $(P_\alpha = P_L)$ would not be attained.

Finally, we observe that transient transport barriers of the types found experimentally can be usefully exploited during the approach to ignition to shorten the most critical phases of the heating cycle.

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6 Ignition criteria: natural and optimal densities for ignition

We may refer to a natural density $\langle n_N \rangle$ for an ignition experiment. The value $\langle n_N \rangle$ is the density at which a pure ($Z_{eff} = 1$) 50:50 D-T plasma ignites most readily for the nominal plasma parameters. It is a characteristic property of a specific machine, considering the achievable plasma size and shape, magnetic field, plasma current, and auxiliary heating power, and it can also be defined for each operating scenario within a given machine design. Since there are maximum and minimum density values allowing ignition in a given experiment, determined by a balance between radiation power loss, available heating power, and energy confinement (and other factors, see [1,2]), there is also the possibility that n_N may not exist for a given case. When it does, it indicates the best possible ignition performance for that device, since the required heating and plasma confinement at n_N are the minimal requirements. Thus, n_N provides a measure of the potential plasma performance at the design operating conditions, even though these may be very different from the ideal conditions used to determine it. Every real plasma will be at least slightly contaminated and thereby suffer degraded performance. The difficulty of achieving the desired operating parameters depends on the degree of improvement needed in the heating power, confinement, etc., over the ideal level, based on the expected degree of contamination (Z_{eff}), which is a sensitive function of density.

The natural density is estimated using a 1 1/2 D transport simulations [2,3]. Different operating regimes [4] in Ignitor can be used as an example. Table 6.1 shows results for the volume-averaged $\langle n_N \rangle$ for the reference scenario at the full field and full current and for a reversed shear, improved confinement regime at reduced current, both with relatively flat density profiles.

Table 6.1: Natural Densities for Ignitor (volume-averaged)

	$\langle n_N \rangle$ $(10^{20}\,\mathrm{m}^{-3})$	T_{i0}/T_{e0} (keV)	B_T (T)	I_p (MA)	$\bar{n}_G$ $(10^{20}\,\mathrm{m}^{-3})$
Reference	5	$\lesssim$ 15/15	13	12	17.3
Rev. shear	3	17/19	12	7	10.1

The reference scenario, based on Refs. [3] and [5], shows ignition at low central temperatures, $T_{e0} \simeq T_{i0} \sim 12-15\,\mathrm{keV}$, with confinement slightly above L-mode and ohmic or almost entirely ohmic heating. These results (actually obtained for $Z_{eff} \simeq 1.2$, but very similar to those for which $Z_{eff} = 1$) are close to expected operating conditions and have also been arrived at by independent evaluations [6,7]. In comparison, the reversed shear case at 12 T and 7 MA [4,8], has approximately $\langle n_N \rangle \sim 3\times10^{20}\,\mathrm{m}^{-3}$ at maximum $P_{Aux} \simeq 8\,\mathrm{MW}$ during the current ramp, assuming a maximum enhancement factor $H = 2.5-3$.

The value of $\langle n_N \rangle$ varies roughly [9] like the Greenwald line average density limit [10], $\bar{n}_G \propto I_p/(\pi a^2)$, as a function of plasma size and plasma current, although with a somewhat weaker dependence on the minor radius. It occurs because the density rise and the rate of current penetration are inter-dependent, the magnitude of the density affecting the local temperature for a given heating rate, and the local temperature in turn determining the resistive diffusion rate of the current. The relationship is less direct when substantial plasma fuelling occurs after the current ramp.

The actual optimal working density for an experiment can be estimated by considering different rates of growth of the plasma density during the current ramp phase [7], and different density profiles at realistic values of Z_{eff}. A further optimization is carried out by varying the reference plasma temperature at ignition and we optimize the plasma pressure in such a way that the peak electron temperature lays in the range $T_{e0} \simeq 10-15\,\mathrm{keV}$. The relevant reactivity function $\langle \sigma \mathrm{v} \rangle_F / T^2$ is represented in Fig. 6.1. Finally we observe that, while looking for the proper plasma parameters near the optimal values of plasma density and temperatures at ignition, we have found the pressure profiles obtained for relatively different conditions to be quite similar for the same class of transport coefficients [7], as illustrated by Fig. 6.2.

It is also important to consider the implication associated to the empirical density limit that has been consistently observed in a wide range of experiments

$$\bar{n}_{lim} = \kappa\langle J\rangle = \frac{5}{\pi}\frac{B_T}{R_0}\kappa\left(\frac{q_\psi}{q_E}\right)\frac{1}{q_\psi}$$

Without compromising the best possible values of $n_0\tau_E$ that can be realistically reached, it is clearly advantageous to maximize both $B_T\ /\ R_0$ and $\kappa q_\psi\ /\ q_E$, where $q_E \equiv (a/R_0)^2\kappa I_M\ /\ I_p$, $I_M = 5R_0B_T$, and q_ψ is the edge safety factor. In practice, because of vertical stability considerations, we consider 1.85 to be an upper limit for κ and likewise, because of macroscopic stability of the plasma column, we take 3 as a lower limit for q_ψ. Thus the ratio q_ψ / q_E can be maximized by adopting the lowest aspect ratio that is practically possible. This kind of optimization considerations has, in fact, had a key influence on the Ignitor design.

We note also that Ignitor is designed to carry out a series of meaningful experiments on plasma regimes that are outside its natural parameter region for ignition. In particular, regimes where the ratio of the alpha particle energy density to the plasma thermal energy density is higher than that corresponding to the parameters of Table 3.2 can be produced. In fact, the adopted auxiliary heating system can be applied to lower density regimes in which the plasma temperature can be raised to considerable higher values than those listed in Table 3.2, to attain the desired ratios of the alpha particle pressure relative to the plasma pressure. We observe that the factor whose variation is particularly interesting to investigate is the gradient of the alpha particle pressure that is characteristically steep in Ignitor.

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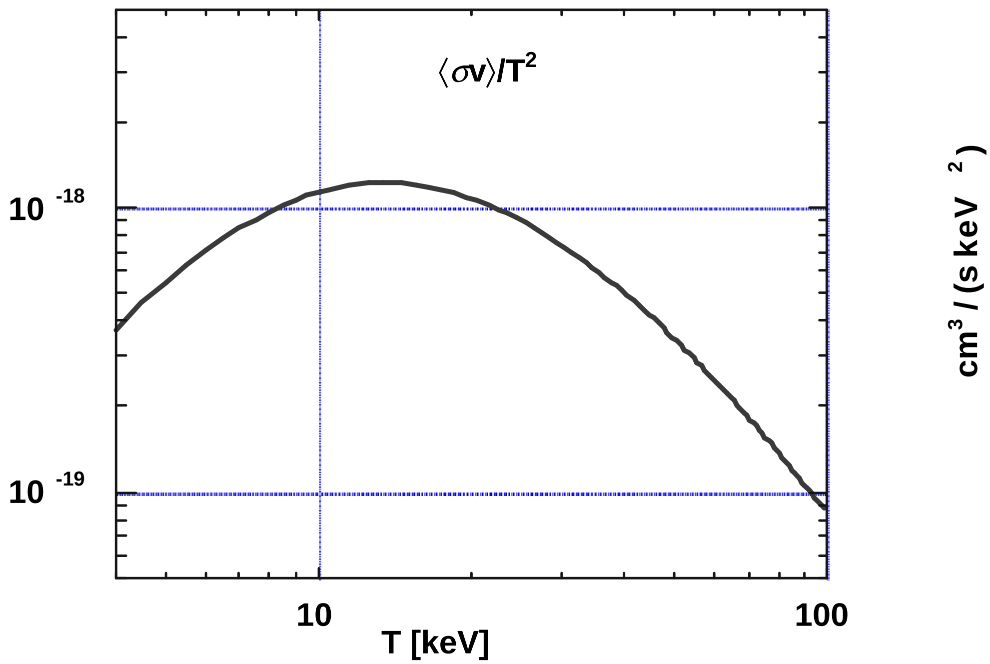

Fig. 6.1. Function <σv>/T^2 whose maximum is at $\simeq$ 14 keV.

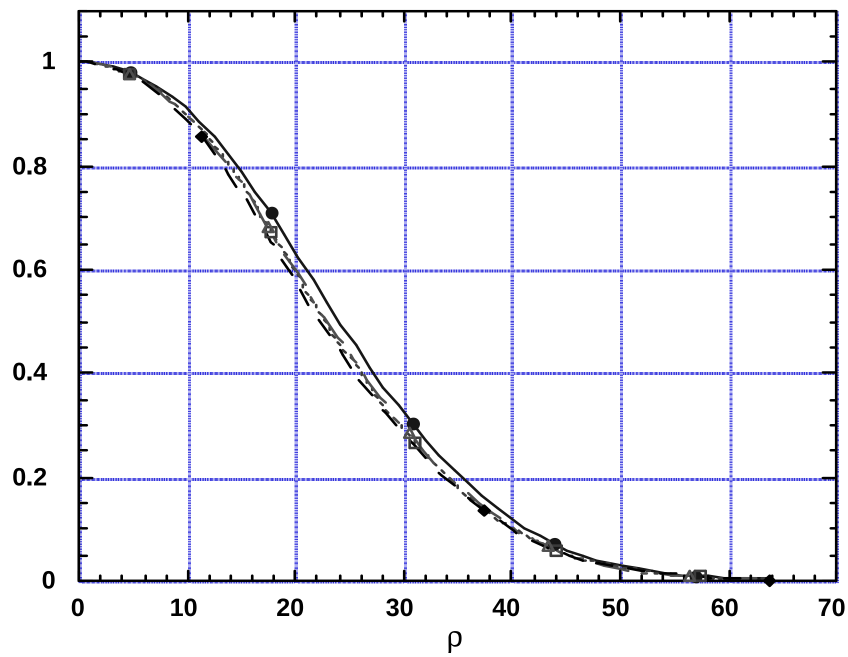

Fig. 6.2. Normalized electron pressure profile at ignition corresponding to different ignition times and different density profiles.

7 Confinement issues

7.1 Confinement and transport models

The difficulty of predicting even the global level of plasma transport (energy and particle confinement) for a given plasma configuration with a good degree of reliability is one of the main problems encountered in choosing the parameters for an ignition experiment. High field experiments at high density require the least extrapolation from the available experimental database, but still lie outside the range of existing experimental data.

Some basic considerations for predicting transport and performance in ignition experiments can be made. First, 0D (global, volume-integrated) steady-state models are not sufficient to predict ignition, as they give only a rough idea of global power requirements. They provide a functional relationship between input power and loss for a given burning level, but do not predict the optimal point for operation, and say little about the possibility to achieve a given operating point in practice. At a minimum, time-dependent, 1 1/2D transport simulations are needed for performance prediction and to envision the relevant control systems, because the energy balance is intimately tied to the plasma profiles (including $q(\psi)$ and current density) and therefore to plasma stability [1,2,3,4].

Second, frequently considered global scalings for energy confinement time are based on a set of criteria and an experimental database [5] that have been chosen in view of a particular design, the ITER EDA [6], whose requirements are different from those of high field designs. One result is the ITER89-P scaling for the L-mode confinement time for which the energy confinement time, τ_E, degrades with the total heating power roughly as $\tau_{E,L} \propto 1/P_H^{1/2}$, or even more strongly [5,6]. An important question is whether different selection criteria, more suited to high field ignition conditions, would yield different results. In fact, as pointed out in Ref. [7], if we apply the so-called "Kaye-All-Complex" scaling to high field ignition experiments, this gives drastically better values for τ_E than the ITER89-P scaling. Since the database used to propose both of these scalings include experiments involving a large diversity of plasma regimes and conditions, it is not surprising that they should not be unique nor give the same extrapolations.

In fact, such criteria can lead to different confinement predictions. A case can be made that the degradation of τ_E with the heating power P_H ceases above a certain power level [8]. This is the prediction of the Coppi-Daughton effective thermal diffusion coefficient [9,10] suggested by a relevant series of experiments carried out by the Alcator C-Mod machine. The corresponding scaling for τ_E was in fact derived from the experimentally observed behavior of β_p in ohmic and RF-heated discharges [8], where $\beta_p \simeq$ constant ($\simeq$0.25) for OH heating, while with additional ICRH, β_p increases linearly with P_{ICRH}. Thus, the resulting τ_E does not have a power law dependence on the plasma parameters, but an offset relation that suggests that the confinement ceases to degrade with heating power above a certain power level,

$$\tau_E \simeq 0.031 R q_E^{2/3} I_p \left(1 + f_3 \frac{I_p V_0}{P_H}\right)\left(\frac{d_i}{a}\right)^{1/2}\left(\frac{\omega_{pe}}{\Omega_{ce}}\right)^{1/3} \qquad (1)$$

in MA, MW, and mks units. Here the coefficient is $f_3 \simeq 1.4(R_0/5a)^{1/2}(R_0/20d_i)^{1/2}$, where $d_i = c/\omega_{pi}$, ω_{pj} is the plasma frequency for species j, Ω_{cj} the gyrofrequency, q_E the "engineering" safety factor, $q_E = 2\pi a^2 \kappa B_T / (\mu_0 I_p R_0)$. The characteristic voltage $V_0 = 0.18 \frac{T_e}{e}\left(\omega_{pi} \frac{c^2}{\omega_{pe}^2} \frac{\nu_e}{\mathrm{v}_{the}^2}\right)^{2/5} \approx 1$ for typical tokamak parameters, is a weakly increasing function of density. All numerical coefficients were determined from Alcator C-Mod data. The resulting expression for τ_E was then shown to fit the global energy confinement times of a specific subset of the ITER L-mode and OH database (as it existed in 1997), with no additional free parameters. The subset was chosen to be more applicable to high density, high field experiments than the general ITER database. It consisted of all the data points satisfying

- OH or L-mode
- clean: $Z_{eff} < 2$

- $T_i \simeq T_e$: $0.7 < W_i / W_e < 1.3$
- mostly thermal: $W_{th} / W_{tot} > 0.7$
- steady state: $(dW/dt)/P_H < 0.1$.

Using a volume-averaged β_p gave excellent results, with a RMS error of 13.1%, compared to 23.6% for the 1996 ITER96 L-mode scaling [8] restricted to these cases. (The ITER96 scaling had a lower error than the original ITER89-P scaling.) Only 7 of the 14 machines represented in the full ITER database appear under these criteria. A thermal diffusion coefficient with an appropriate dependence on the flux surface was also derived and shown to fit a wide variety of steady state ohmic and RF-heated L-mode discharges from Alcator C-Mod.

We note that, as was clearly shown by the series of experiments carried out with the Alcator C machine, the quality of confinement in ohmic heating discharges is not unique. For instance, peaked density profiles resulting from the injection of pellets have been observed to produce considerably higher energy confinement times. The Alcator C-Mod ohmic discharges on which the scaling (1) is based were non-optimized. No attempt was made to improve the confinement time by any means. Therefore, Eq. (1) should be considered too pessimistic in trying to assess the confinement time in Ignitor.

The radial form of a transport coefficient is important for predicting ignition, which is a strongly dynamic and non-local process. Then we observe that, even if nearly stationary conditions are accurately described by a given transport coefficient, this may give poor results under dynamic conditions. In fact, numerical transport simulation consistently indicates that a coefficient that nearly preserves the temperature profile shape (“profile consistency” [11]) is required to fit many present-day experiments, especially their transient phases, as well as having a strong effect on ignition predictions. For example, the CD97 coefficient described in Refs. [9,10] does not work as well for transient conditions, including Alcator C-Mod current ramps and ignition simulations, because its strong dependence on the plasma pressure gradient tends to produce an artificially steep gradient at mid-radius. In particular, a simple profile-consistent coefficient, such as the original CMG (Coppi-Mazzucato-Gruber [12]) scaled to match a desired global confinement, has been shown to be quite adequate [3,4].

The numerical transport codes that have been employed mostly are BALDUR (fixed boundary) [13], TSC (free boundary) [14], and JETTO (free boundary) [15].

7.2 Effective Thermal Diffusivity

As indicated earlier, the possibility to reach ignition conditions depends on, among other factors, the *radial* profiles of both the electron and the ion effective thermal diffusivity. The models of these parameters that appear to simulate best existing experiments, both under stationary and dynamic conditions, have in common the fact that they increase toward the edge of the plasma column. This circumstance is, in fact, an aspect of the "profile consistency" argument [12,16]. Thus one of the problems on which an ignition experiment will have to shed light is whether the fusion reaction products will be scattered by collective modes to deposit their energy in the outer region where the thermal conductivity is relatively high.

High magnetic field and plasma density experiments have shown that extremely low thermal diffusivities can be produced in the central part of the plasma column. In particular, we refer to the experiments carried out by the Alcator C machine where confinement times slightly exceeding 50 msec were obtained with a plasma radius of 16 cm, peak plasma densities $\simeq 2 \times 10^{21}$ m^{-3}, $T_e \simeq T_i$, and $Z_{eff} \simeq 1$. For these plasmas we may estimate a corresponding mean effective thermal diffusivity as

$$\bar{\chi}_E \simeq \frac{a^2}{4\tau_E} \simeq 0.13\,\mathrm{m^2/sec}$$

We note that the poloidal magnetic field corresponding to a current of about 750 kA in this case is slightly below 1 T.

The mean effective thermal diffusivity that in Ignitor would correspond to a confinement time $\tau_E \simeq 0.5$ sec can be estimated as

$$\bar{\chi}_E \simeq \frac{\bar{a}^2}{4\tau_{\mathrm{E}}} = \frac{0.47^2 \times 1.83}{4 \times 0.5} \simeq 0.2\ \mathrm{m^2/sec}\ .$$

An interesting point is to try to anticipate the confinement times that Ignitor should attain under predominant ohmic heating conditions. For this we note that ohmic heating regimes, where $T_e \gtrsim T_i$ have two salient characteristics:

i) the observed loop voltage values lay in a small range e.g. 1-1.5 volt

ii) profile consistency [16].

These are the basis of the CMG diffusion coefficient that scales as

$$D_\Omega \sim \frac{I_p V_\parallel}{qnT_e R} \propto \frac{a}{R}\frac{V_\parallel}{\beta_p q}\frac{1}{B_p}$$

where $V_\parallel$ is the loop voltage. Therefore we may argue that, for $\tau^\Omega \propto a^2 \kappa / D_\Omega$, and taking $\tau^\Omega_{Ac} \gtrsim$ 50 msec for the case of Alcator C, we would obtain a relatively large value for τ^Ω_{IGN}, assuming comparable values of $V_\parallel / (\beta_p q)$ for the two cases. This leaves room for a large possible degradation factor when the increased input power associated with α-particle heating is taken into account. We note that in the Alcator C experiments the estimated transport of ion thermal energy due to collisional effects was significant. Therefore, anticipating the confinement time without distinguishing between the nature of the electron and the ion transport, even though this is a common practice, is certainly debatable. In addition, when considering the best ohmic plasmas, which can be produced in Ignitor, the bremsstrahlung power loss becomes important at the highest densities and it should be subtracted from the heating power in order to evaluate τ^Ω_{IGN}.

In this context we may refer to the “Sheffield plot” (Fig. 7.1) from which is clear that the high density plasmas with $T_e \simeq T_i$ produced by high field machines have the best confinement quality. The value of $2\beta\tau_E / \bar{a}^2$ that can be extrapolated for Ignitor is about 0.2 s/m^2 and this would give a long confinement time even assuming a factor 2 degradation.

A significant example of the observation that peaked density profiles like those resulting from pellet injection lead to enhanced confinement times is provided by the $n_0\tau_E$-record discharge produced by the TFTR machine with $T_e \simeq T_i$ and ohmic heating only [17]. We note that the relevant poloidal field $\bar{\bar{B}}_p$ was about 0.45 T, corresponding to a poloidal field pressure about 1/60 of that expected at 11 MA in Ignitor.

The FTU machine also obtained an improved confinement regime with an enhancement factor of about two following the injection of multiple pellets in ohmic plasmas [18], resulting in higher central densities and more peaked density profiles. The analysis of these discharges showed that the ion thermal conductivity was reduced to nearly neo-classical values, whereas the electron thermal conductivity was essentially suppressed in the region inside the $q = 2$ surface.

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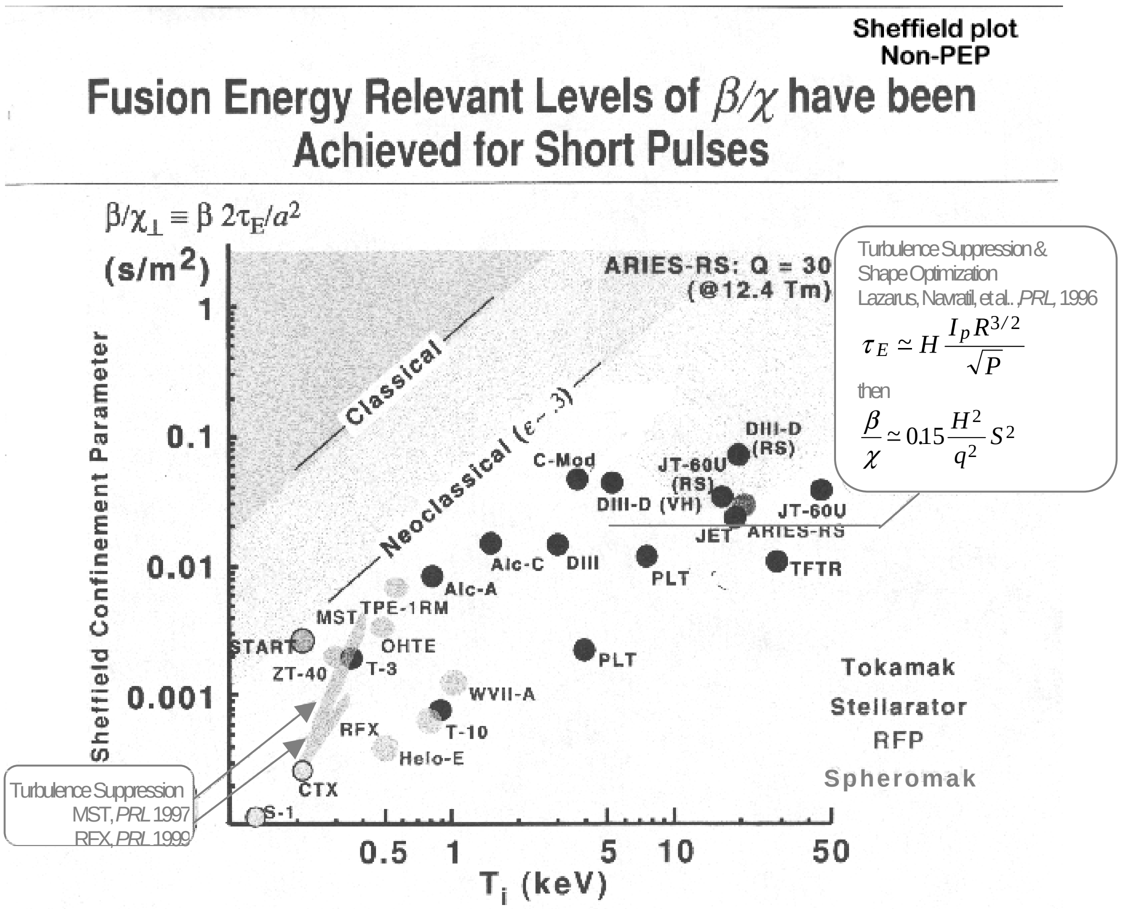

Fig. 7.1 Graph produced by J. Sheffield.

8 Additional considerations

8.1 Open questions on transport

Other open questions about thermal transport in burning plasmas remain. For example, the heating of the plasma due to collisional slowing down of the charged particles produced in fusion reactions is isotropic in velocity space and axisymmetric in real space, with its magnitude centrally localized in the plasma. Does it then cause degradation of confinement time with increasing input power in the same way as most existing methods of injected heating, which are anisotropic in velocity space, non-axisymmetric in space, and often concentrated off-axis? This empirical rate of degradation with power exerts perhaps the most crucial influence on current designs for ignition experiments and potential reactors. Evidence that some heating methods, such as ECRH, do not degrade confinement in this way [1] will receive our continued attention.

Another question is whether the current density redistribution is solely due to collisional effects even in the high electron temperature regimes considered or whether collective modes, driven for instance by the current density gradient, may have an effect on it. In connection with this we note that Ignitor is designed to have record high current density gradients.

A further consideration for fusion burning plasmas is that the electron thermal transport is important, unlike present-day lower density, mainly ion-heated experiments with $T_i > T_e$, that are dominated by the ion thermal transport attributable to toroidal ion temperature gradient (ITG) modes. Relatively little has been done for electron transport, even for global scalings. The theory of electron transport processes has yet to be compared satisfactorily to the experiments. Moreover, the connection between fluctuations and transport is much more difficult to simulate numerically than for the ions. On the other hand, it is known from experiments with very high densities and peaked density profiles such as those carried out by the Alcator C machine, that the electron thermal energy transport can be greatly reduced. We note that the peak electron temperatures which Ignitor has been designed to reach at ignition have already been attained in present experiments and the values of the required energy confinement time are well within the range of those already established.

8.2 Enhanced confinement regimes

A number of improved confinement regimes have been established in the course of experiments with advanced machines, starting from the enhancement of ohmic confinement as a result of pellet injection (predicted on the basis of the characteristics of ITG modes) found in Alcator C, the related PEP mode in the presence of auxiliary heating identified later in JET [2], the H-mode found in ASDEX [3], up to the more recent Reversed Shear (or Negative Shear) mode of operation [4]. As it was observed in Section 2, frequently these improved confinement regimes are either transient in time and/or characterized by significant non-Maxwellian particle distributions (i.e., the “Hot-ion mode” in JET). An increase in Z_{eff} is another problem often associated with enhanced energy confinement regimes, since particle confinement also improves along with energy confinement, leading to impurity levels in the plasma that are unacceptable for an ignition experiment. Exceptions are the ELMy H-mode (ELMs are edge localized modes), and the so-called Enhanced D_α (EDA) H-mode, routinely observed on Alcator C-Mod [5,6] but also on other devices. These regimes rely on edge-localized phenomena (semi-periodic MHD activity in the former and high edge recycling of neutrals in the latter case) to maintain steady-state conditions in the plasma core. Constant density and radiated power can be sustained for long times, with relatively low impurity accumulation and good energy confinement times τ_E, relative to corresponding ELM-free H-modes. The physics governing the transition to the H-modes is still poorly understood, and the minimum amount of power required can only be estimated, relying upon statistical extrapolation from a range of very different experiments. The latest proposed scaling law for the threshold L-H transition power P_{L-H}, derived from D-D experiments, is [7]:

$$P_{L-H} \simeq 0.054 \bar{n}^{0.49} B_T^{0.85} S^{0.84},$$

where $\bar{n}$ is the line average density in 10^{20} m^{-3}, B_T the toroidal magnetic field in tesla, S the plasma surface in m^2 and P_{L-H} is in MW. According to this expression, approximately 23 MW, without taking into account a possible favorable effect of the isotopic mass, of input power is necessary for Ignitor at full parameters, less during the current ramp.

The Ignitor strategy is to reach the desired value of the ignition parameter $n_0 \tau_E T_0 \sim 6 \times 10^{21}$ m^{-3} sec keV at low temperature (~ 12 keV) and high density (~ 10^{21} m^{-3}). Therefore, only a relatively modest $\tau_E \sim 0.5$ sec is required, which is compatible with the current predictions for an L-mode type of confinement, when the dynamical nature of the ignition process is properly taken into account. Of course, it is not known exactly what effect the presence of substantial alpha particle heating will have on global confinement and whether it will be similar to other forms of externally injected heating or not. It is often argued that the Ignitor design, in the absence of a divertor and with limited additional power, does not allow the conventional access to an H-mode regime, so that there would be no margin to achieve ignition if the ohmic confinement alone falls below the expected values. It should be observed that the H-mode is not the kind of enhanced regime most favorable for ignition, since it is characterized by very broad and flat density profiles. Therefore the possibility of achieving the H-mode in Ignitor is being investigated as an interesting physics option. Instead, almost any form of Internal Transport Barrier would be more helpful, even if transient. Ignitor is very well suited for a Reversed Shear mode of operation, at lower current and higher q_ψ, aided by a modest amount of additional ICRF power [8]. This and a pellet injector are in fact included in the Ignitor design. In any case, given the possibility of operating with an X-point or two "near X-points" configurations, H-modes should be possible to achieve.

An equilibrium configuration at I_p = 10 MA, B_T = 13 T, with a double X-point laying just outside the first wall, can in fact be produced by the designed poloidal field system, as shown in Fig. 8.1. The compatibility of this mode of operation with the thermal loads on the first wall tiles, which will be the subject of detailed analyses is not expected to be a problem.

8.3 Internal *m* = 1/*n* = 1 modes and their characteristics

In a toroidal plasma configuration with $q_0 < 1$, the excitation of global *m* = 1/*n* = 1 internal modes can limit the maximum plasma pressure gradient that can be confined in the central part of the plasma column for given values of the magnetic field components [9]. In a tight aspect ratio configuration, the amplitude acquired by the coupled *m* = 2 harmonic, which can involve a considerable fraction of the plasma volume, is an additional concern. Moreover, for low-*q*, relatively high-β_p operation, these modes may nonlinearly destabilize ballooning modes [10] and/or induce seed islands for the growth of neoclassical tearing modes [11]. In an ignition experiment the plasma heating, clearly, depends strongly on the central plasma pressure. Understanding the conditions for stability of *m* = 1/*n* = 1 internal modes is therefore an important albeit complex subject, and one for which direct evidence obtained with a real burning plasma experiment might ultimately be required. For the case of Ignitor, the stability of *m* = 1 /*n* = 1 internal modes has been investigated using in particular a combined analytical-numerical approach; the ideal MHD computations have been performed starting from the analyses given in Refs. [9,12] (for details, see also [13]). Fast alpha particle effects [14] as well as finite resistivity, finite "drift" frequency and finite ion Larmor radius effects have been considered. For Ignitor equilibria that are stable or weakly unstable according to ideal MHD theory, alpha particle and finite diamagnetic frequency effects stabilize the relevant resistive modes. To insure ideal MHD stability, a procedure for control of the steepness of the plasma pressure profile and of the shape of the *q*-profile is certainly desirable. In fact, one of us (P.D.) has found that there exists a class of "shoulder" *q*-profiles that have superior ideal MHD stability. We also note that for control of this profile, early application of moderate ICRH power during the current ramp may be an important option [15].

Finally, we observe that the typical values of β_p for the ohmic plasmas produced by the Alcator C-Mod machine are around 0.25, similar to that expected in Ignitor at ignition. The associated sawtooth activity, which is always present when $q < 1$ inside the plasma, does not involve large scale temperature oscillations. These are instead consistently observed in plasmas with ICRH injection that are characterized by higher β_p and are in the H-mode regime [16].

8.4 Control issues: density profiles and burn conditions

The prediction and control of the density profile at high densities is another important transport and edge plasma physics problem for ignition experiments. The basic shape of the density profile cannot be reliably predicted from present knowledge. Peaked density profiles are more favorable for ignition, although the level of degradation with flatter profiles is relatively small, as long as the total number of particles remains roughly the same.

The question of the degree of profile control (peaking) by pellet injection, which translates to the question of the penetration of the pellet particles into the plasma, has been a subject of strong interest, that included an experimental effort [17], within the Ignitor program. Control of the plasma edge density during startup and steady state is also important, since it regulates the current penetration rate as well as being related to the edge temperature. A balance must be struck – high edge density improves impurity screening from the main plasma, but may be less beneficial for other processes, such as plasma heating and/or stability. (High edge densities result in relatively lower edge temperatures, which speed up the rate of the edge current penetration and have an influence on the central plasma temperature, which tends to be reduced).

Transport simulation readily demonstrates that precise time-dependent burn control through variation of the bulk ion density source is *not* possible in general, since particle confinement times τ_P are generally longer than energy confinement times τ_E. Short-time-scale sensitivity to the fuel-ion particle source rate requires that the confinement τ_E be marginal relative to that needed to maintain the desired level of burning, or that the burning rate is high enough that a strong source of fuel ions is required to sustain it. A generalized form of burn control by specifying the concentration of tritium relative to deuterium in a discharge can always be used. Much better control is possible by operating in a slightly sub-ignited state that is driven by a small amount of externally supplied heating.

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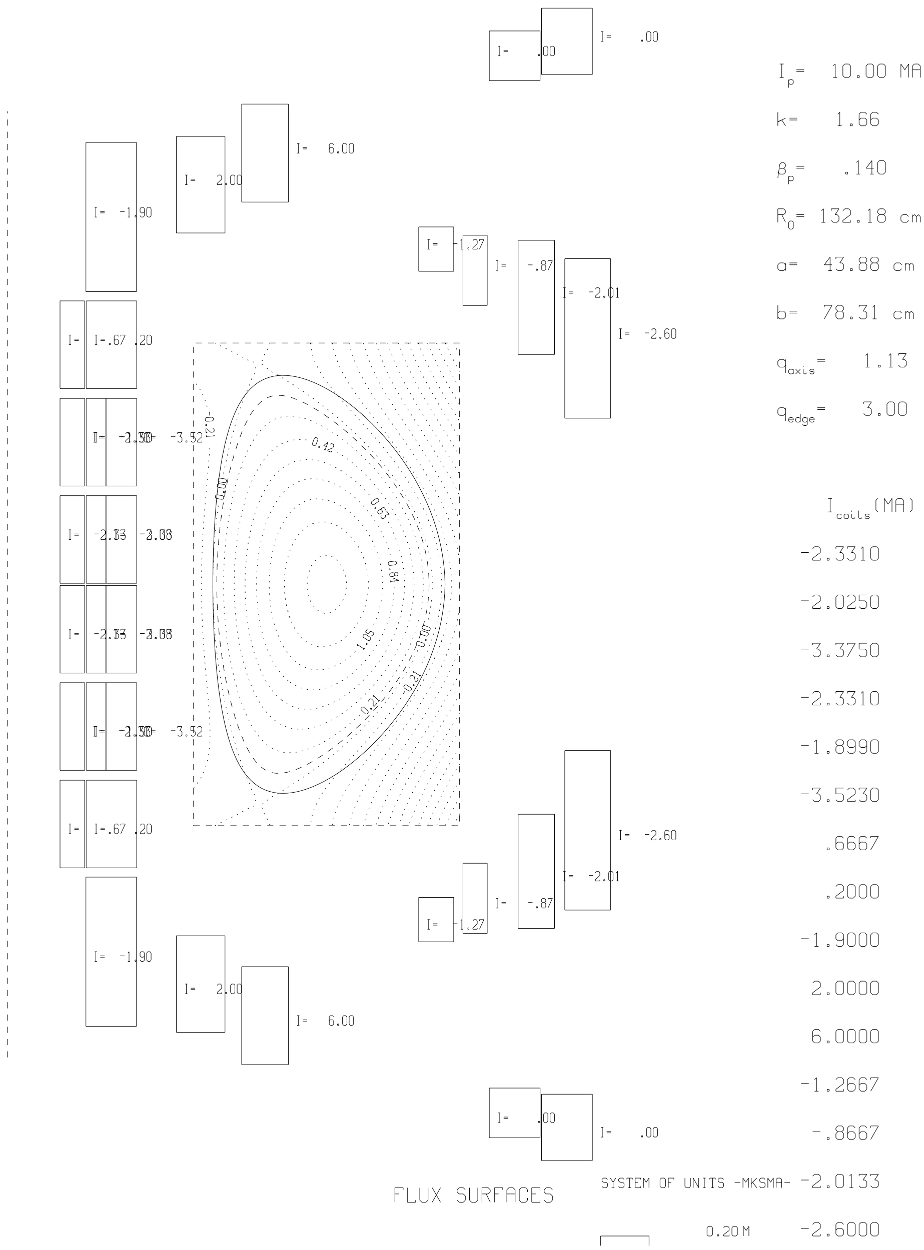

Fig. 8.1. Example of an equilibrium configuration for 10 MA and 13 T, as evaluated by the EQUISL code, with a double X-point laying just outside the first wall.

9 Plasma-surface interaction issues

Power is exhausted from magnetically confined plasmas either in the form of electromagnetic radiation or as energetic particles. An important concern is that of reducing the power load on the physical walls exposed to the plasma, where impurities can be produced by sputtering and/or evaporation. Plasma-surface interactions need be controlled in order to maintain a clean plasma core, i.e. with a reduction in the production of intrinsic impurities and their screening from the plasma core (Fig. 9.1), and to ensure a wide dispersion of power over the walls. Thus far, two solutions have been envisaged to deal with these issues:

i. The divertor concept, where the magnetic field lines at the periphery of the plasma are “diverted” to a location separate from the main plasma. The remote location from the plasma heat source allows the plasma temperature adjacent to material surfaces to be reduced, hence reducing the sputtering yield of the incident plasma ions. Although impurities that are generated in the region are potentially screened from the main plasma, it is not presently clear whether the divertor has indeed led to cleaner core plasmas over those in limiter devices [1]. One negative feature of the divertor is that it tends to “focus” power over relatively small areas, i.e. the divertor plates.

ii. The series of experiments on record high density plasmas carried out by the Alcator machines, with appropriate limiters, that produced unexpected high degree of purity have opened the way to another concept, where most of the plasma energy is lost by impurity radiation at the edge of the plasma column. In this case the conversion takes place in a rather narrow layer and does not affect the global energy confinement time. As shown in Fig. 9.1 the impurities are, in effect, hindered from entering the plasma column in the regimes with the highest densities. In particular, the “cold radiating plasma mantle” idea is supported by the transport analysis given in Ref. [2]. Later experimental observations (in both limiter and divertor machines) have demonstrated the possibility of operating with a radiative mantle which can dissipate up to 90% of the total power lost by the plasma without energy confinement degradation [3,4].

After weighting different alternatives, Ignitor has adopted the solution with limiter, on the basis of the second concept, with molybdenum tiles as the first wall material. This solution is in fact suitable to the requirements of plasma-wall interaction control in high density plasmas. The main relevant considerations are:

1) Plasma purity: minimal contamination and excellent confinement properties in high-density plasma regimes have been observed consistently, since the first Alcator and FT experiments, in a large variety of experiments that lately have included the reversed field pinch RFX. A scaling law relating plasma purity, radiated power, and machine dimensions has been derived from a significant database of toroidal confinement experiments [5]. The result from this scaling law is given in Fig. 9.2, using the reference Ignitor parameters. According to this, Ignitor is expected to radiate most of the input power, from the edge, while $Z_{eff} \lesssim 1.2$. The relevant fraction of molybdenum ions relative to the electrons (Z_{Mo} = 42, average charge state $\langle Z \rangle_{Mo}$ = 30) that is estimated does not compromise the possibility to reach ignition.

2) Thermal loads: these have been calculated assuming an ideal continuous first wall, under the conservative hypothesis that only 70% of the input power is radiated. Under ignition conditions, a maximum thermal load of 1.8 MW/m^2 is found (see Fig. 9.3) when “physiological” plasma movements of ± 1 cm around the equilibrium configuration are considered (the expected average heat flux is $\lesssim$ 0.7 MW/m^2).

3) Energy confinement time: radial profiles for the radiated power have been calculated by means of a self-consistent model [6], that couples the core of the plasma column with a single impurity species and the scrape off layer (SOL). The resulting radiating profiles show that 40% of the radiative losses are localized in the outer region of the main plasma and 40% in the SOL. These radiative regions are close to the first wall surface, and do not affect the confinement conditions of the core of the plasma column.

The value of the edge density in Ignitor, $n_a \simeq (2 - 3) \times 10^{20}$ m^{-3}, has been estimated from a model that assumes edge fuelling and a simple edge transport model [6], which gives a very good fit to a wide range of limiter experiments. Figure 9.4 shows the values of n_a from these experiments and the estimated value for Ignitor. The edge temperature has been derived by an energy balance between the total power, the radiated power and the power transported to the limiter, with the assumption of no temperature gradient in the SOL. Temperatures $T_a \simeq 35 - 60$ eV at the last closed flux surface are expected for core radiated powers between 10 and 25 MW.

The Poloidal Field Coils system of Ignitor can be adapted to introduce an X-point of the magnetic configuration within the first wall, as stated earlier, in order to access the H-mode regimes in a conventional way. However, there are several disadvantages associated with an X-point configuration, a major one being that of a significant reduction in the plasma cross sectional area. Consequently, the plasma current which can be produced would be significantly reduced. In addition, the introduction of divertor strikepoints, as mentioned above, tends to focus the plasma power onto a relatively small area on the first wall. Most important, the flat density profiles that characterize the H-mode regimes are less desirable, taking all factors into account, than the peaked density profiles in view of obtaining ignition conditions. Thus, producing an X-point configuration is not considered a priority in the Ignitor design.

Finally, referring to the limiter configuration of Ignitor, we note that sufficient space between the first wall and the chamber at the outboard side has been left for the possible introduction of pumped or vented limiters, noting that such structures may be desirable to improve plasma density control.

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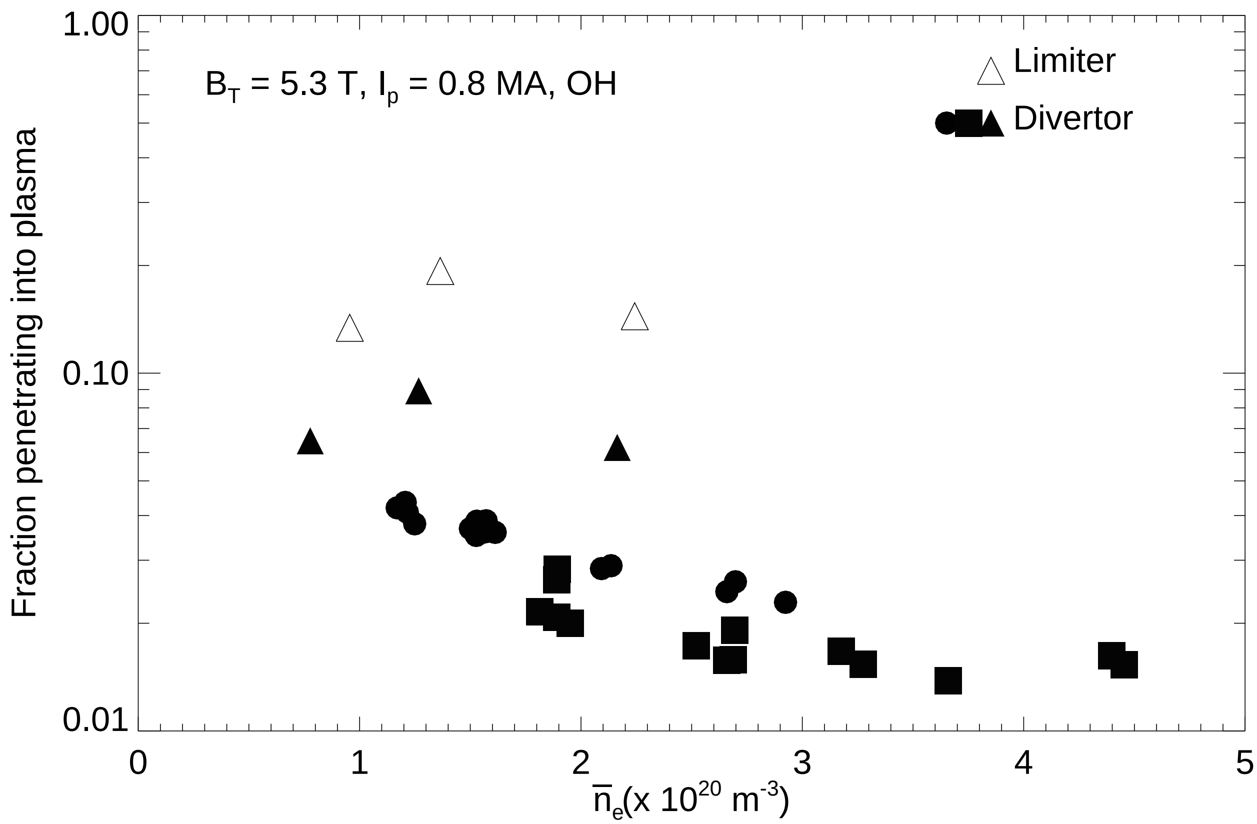

Fig 9.1. Effect of electron density on the plasma screening of Ar impurity ions in experiments by the Alcator C-Mod machine.

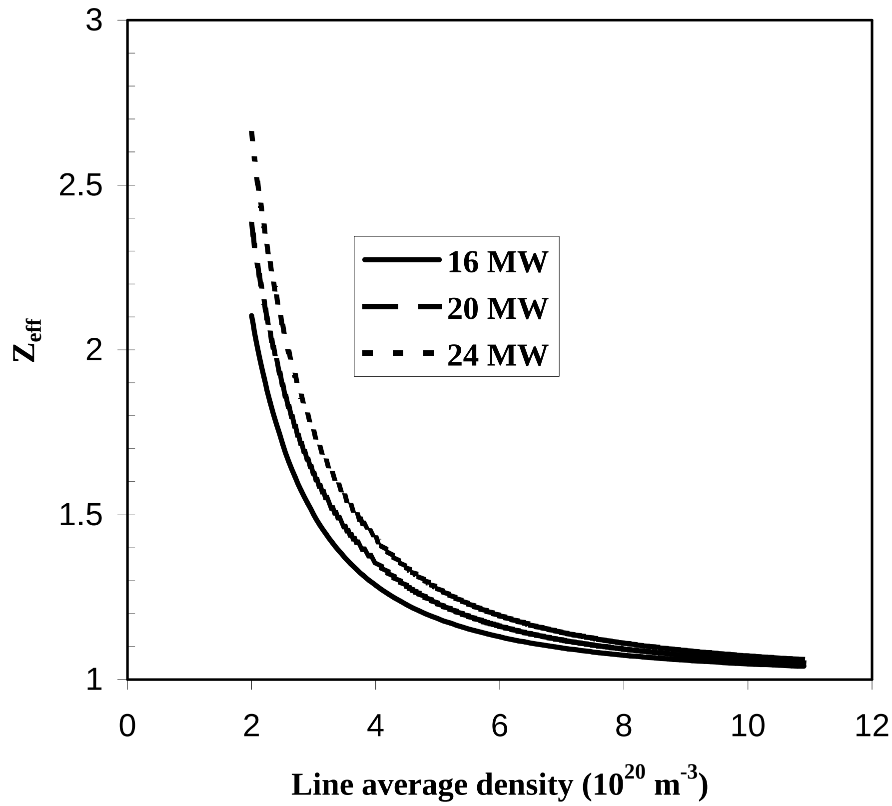

Fig 9.2. Degree of plasma contamination as a function of density according to the scaling of Ref.[5] applied to Ignitor, for different amounts of radiated power.

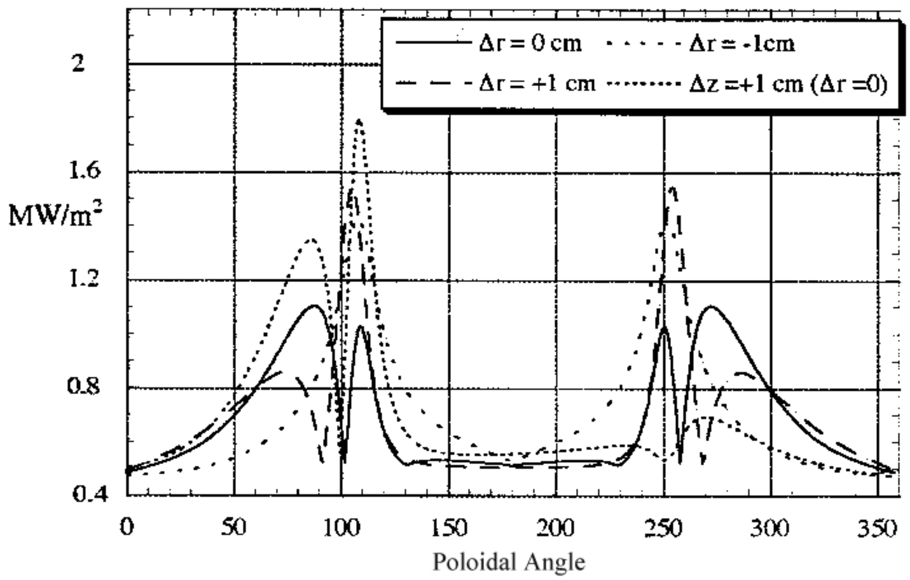

Fig. 9.3. Thermal load distribution under ignition condition for different radial shifts.

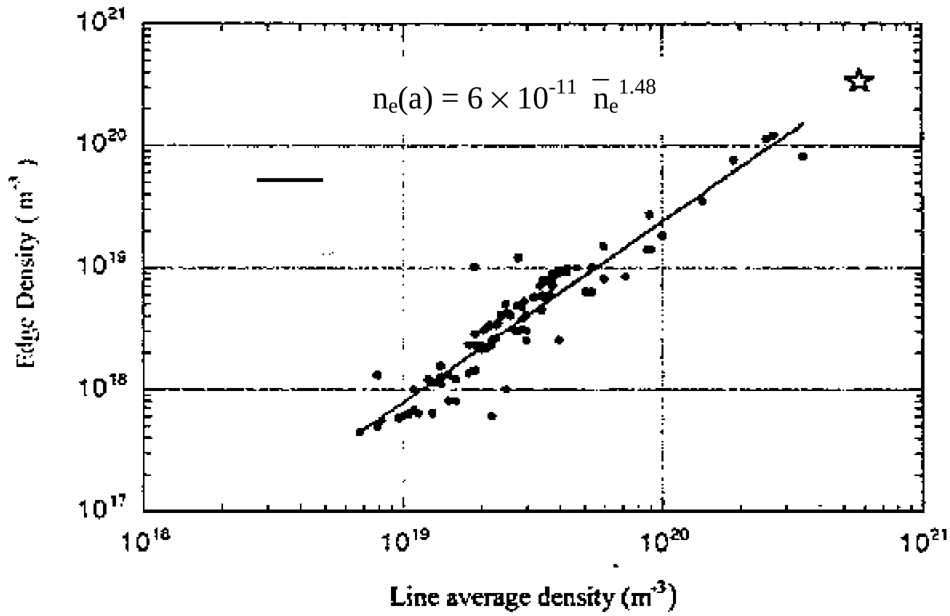

Fig. 9.4. Experimental values of the edge density vs the line average density. The star represents the extrapolated value for the reference Ignitor discharge.

10 Alternative fusion reactions

Given the drastically different conditions under which tritium-poor plasmas can reach ignition compared to D-T, it is of particular interest to explore the physics of plasmas in which D-^{3}He or possibly the D-D catalyzed reactions play an important role. These reactions have their own set of problems, such as the availability of ^{3}He and the attainment of the higher plasma parameters that are required for burning. To begin to explore their possibilities, a D-T burning plasma experiment at high field and plasma densities, which can be much closer to the required parameters than present-day experiments, is particularly attractive. In fact the density limit in high field experiments such as Ignitor, is well above the optimal density for D-T ignition but is suitable to the higher densities required for D-^{3}He burning. As a start, Ignitor can allow initial studies at the level of approximately 1 MW of power in charged particles from the D-^{3}He reaction in a mostly D-T plasma [1,2]. In addition, the technological feasibility of an experiment, called Candor, designed to study the ignition conditions of a D-^{3}He plasma, has been investigated with encouraging results. The high field technologies envisioned for this are similar to those adopted for Ignitor, but the configuration of the main magnet systems has been adapted to the need to produce considerably higher plasma currents and longer pulses.

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11 Developments in the Ignitor Program

A detailed engineering design supported by structural and other relevant analyses has been carried out. This has required the involvement of a broad range of expertise and contributions from a spectrum of research institutions and industrial groups. In order to prove the actual technological feasibility of the most difficult components of the machine, full size prototypes have been constructed and a series of relevant tests carried out. The constructed prototypes include a module ($1/24^{th}$) of the toroidal magnet, a C-clamp that is the largest structural supporting component of the machine, one of the innermost central solenoids, a sector of the plasma chamber ($1/12^{th}$), and a segment of the radial mechanical press. The next round of construction activities include the fabrication of the entire system of central solenoids (the so-called "air core" transformer) and of the additional 23 modules of the toroidal magnet system. Each of these two systems can be tested separately, albeit at lower currents than the reference design values.

In addition, the Ignitor program has spurred the development of new structural concepts such as the so-called "bucking and wedging" to optimize the redistribution of mechanical stresses (also included in the ITER EDA design), the system of radial presses providing an appropriate mechanical torque to decrease the stresses in the "central legs" of the toroidal magnet coils (now adopted in the FIRE design), etc. Another example is that of the neutron spectrometer, which was originally conceived for the Ignitor program by J. Källne and has since been constructed independently and utilized successfully at JET.

An in depth review of both the physics and engineering aspects of the Ignitor program has been carried out in the First and Second International Symposiums on Ignitor that were held in Cambridge, MA (at MIT) in November 1998 and in Washington, DC (at the facilities of LBL, University of California) in May 1999. Both of these symposia attracted a strong and active participation of U.S. scientists and the material presented has been documented in two voluminous sets of reports.

A summary of the characteristics of the collective modes that can be excited in plasmas approaching ignitions conditions and of the unknowns that have to be faced in predicting the outcome of ignition experiments at this time have been given by one of us (B.C.) at the 1999 Snowmass Summer Study. In this context we point out that there has been consensus and continuity of opinions concerning the need for and the value of ignition experiments at this time within the scientific community. This is exemplified by the relevant statements included in the document on fusion research issued by the White House (P.C.A.S.T.) in 1995 (Fig. 11.1) and in the conclusions of the 1999 Snowmass Summer Study (Fig. 11.2). The similarity of these statements is self-evident.

President's Committee of Advisors on Science and Technology (PCAST): Report on Fusion Research

p.22 (White House, July 1995)

Producing an ignited plasma will be a truly notable achievement for mankind and will capture the public's imagination. Resembling a burning star, the ignited plasma will demonstrate a capability with immense potential to improve human well-being. Ignition is analogous to the first airplane flight or the first vacuum-tube computer. As in those cases, the initial model need not resemble the one that is later commercialized; much of what would be learned in a tokamak ignition experiment would be applicable both to more advanced tokamak approaches and to other confinement concepts.

Fig. 11.1. Significance of achieving igniton.

Resolutions of the burning plasma subgroup

A. On the question of justification for a burning plasma experiment, the following resolution was adopted unanimously.

"The excitement of a magnetically-confined burning plasma experiment stems from the prospect of investigating and integrating frontier physics in the areas of energetic particles, transport, stability, and plasma control, in a relevant fusion energy regime. This is fundamental to the development of fusion energy.

Scientific understanding from a burning plasma experiment will benefit related confinement concepts, and technologies developed for and tested in such a facility will benefit nearly all approaches to magnetic fusion energy.

There was some discussion that the burning plasma experiment should be an attractive fusion energy device and not just relevant. However the majority chose to adhere to the word relevant (70%).

The issue of transferability and the entire statement regarding frontier physics was voted on and agreed to unanimously.

B. On the question of what constitutes frontier physics in a burning plasma experiment,
the group agreed unanimously to the following.

FRONTIER PHYSICS TO INVESTIGATE AND INTEGRATE IN A SELF-HEATED PLASMA

- **Energetic Particles**
  Collective alpha-driven instabilities and associated alpha transport.

- **Transport**
  Transport physics at dimensionless parameters relevant to a reactor regime (L/ρi scaling of microturbulence, effects on transport barriers...

- **Stability**
  Non-ideal MHD effects at high L/ρ_I^*, resistive tearing modes, resistive wall modes, particle kinetic effects...

- **Plasma Control**
  Wide range of time-scales: feedback control, burn dynamics, current profile evolution

- **Boundary Physics**
  Power and particle handling, coupling to core

(*L/ρi is the system size divided by the Larmor radius.)

Fig. 11.2. From "Burning Plasma Physics"

(1999 Fusion Summer Study, Snowmass Colorado), pag. 6.

12 Summary

The major points driving the design of the Ignitor experiment can be summarized as follows:

- The crucial physics issues related to fusion burning plasmas and potential fusion reactors can only be studied in an experiment capable of approaching ignition.
- The Ignitor experiment takes the most conservative approach to the near term study of the physics of fusion burning plasmas, through an optimal combination of geometrical characteristics, plasma current, and magnetic field. This approach lends itself to important developments that include advanced fuel burning (low neutron yield, e.g., D-^{3}He).
- The Ignitor design has been strongly driven by the physics of ignition. A large amount of original and early work on the physics has been carried out during the design process, that is applicable to all magnetically confined burning plasma experiments. This statement can also be extended to the engineering design of the machine and the technology solutions devised for it.
- High magnetic field, high density plasmas have the most favorable characteristics and expectations for ignition, and are the only ones that, given the present knowledge of plasma physics, allow this goal to be pursued realistically.

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Appendix A. Requirements for a high field ignition experiment

This Appendix summarizes the set of characteristics required for a tight aspect ratio, high field ignition experiment [1].

The combination of high toroidal fields B_T and compact confinement configurations leads to an interlocking set of characteristics favorable for ignition. In particular, high fields and contained dimensions, with optimized vertical elongation and triangularity, allow a relatively large plasma current, toroidal current density, and poloidal magnetic field to be produced simultaneously. (In Ignitor, the mean poloidal field is $\overline{\overline{B}}_p \leq 3.75$ T. Also, there is a large paramagnetic current $I_\theta \lesssim 9$ MA at the low values of β_p where ignition is attained and this increases the central B_T by $\simeq 1$ T.)

It is well known by now that the maximum plasma density which can be confined correlates with the current density (i.e., $n_{Max} \propto I_p/(\pi a^2)$) which in turn is related to the ratio B_T/R. Record high densities have in fact been produced by the Alcators A and C, FT and FTU, and TFTR machines. In particular, Alcator C obtained $n_{e0} \simeq 2 \times 10^{21}$ m^{-3} at $B_T \simeq 12.5$ T. The volume-averaged current density in Ignitor can be as high as $\langle J_\phi \rangle \simeq 0.93$ kA/cm^2. This should allow $n_{e0} \simeq 10^{21}$ m^{-3} without difficulty. Therefore, based on the required $n_0 \tau_E \simeq 5 \times 10^{20}$ s/m^3 for ignition conditions at $T_0 \sim 12.5$ keV for a 50:50 D-T plasma, only a moderate energy confinement time is required.

As a consequence, such plasmas have

- High levels of ohmic heating up to ignition [1] (given the high values of B_p).
- Good confinement of plasma energy and particles (since empirical scalings show that, approximately, $\tau_{E,L} \propto I_p$).
- Good confinement of fast fusion charged particles. ($I_p > 6$ MA will give good central confinement of the 3.5 MeV alpha particles.)
- Low temperature ignition ($T_{e0} \simeq T_{i0} < 15$ keV in Ignitor) with relatively low levels of fusion heating.

- Ignition at low $\beta_{p.}$ Ideal MHD and long wavelength resistive $m = 1$ internal modes are expected to be stable under ignition conditions [2].
- Low fusion power and thermal wall loading.
- Clean plasmas (since Z_{eff} is a monotonically decreasing function of density).

In addition, high field and the ability to ignite at low β gives the capacity for a broad range of operating conditions at less-than-maximum parameters.

In conclusion, these characteristics avoid or reduce the need for

- Injected heating, except to control plasma stability, to extend the operating range, and as a backup to ignition. This avoids serious degradation of confinement before the fusion alpha heating regime is reached and allows the possibility that fusion heating may take over with continuity from the ohmic heating, since it is axisymmetric and isotropic like ohmic heating, and maintain better confinement characteristics than injected heating.
- Divertors, which concentrate the thermal wall loading on small regions. Divertors require an expanded volume inside the toroidal field coils to accommodate the magnetic separatrices, the divertor, and the associated shaping coils. For high field designs, relatively small increases in the size of the coils and the major radius have serious consequences through the cascade of relations: larger $R \rightarrow$ lower $B_T/R \rightarrow$ lower n_e, lower $B_T \rightarrow$ lower I_p and P_{OH}, so that β_p is higher at ignition. The I_p is also lower for given B_T because the necessity of squeezing magnetic separatrices and the divertor inside the toroidal field coils reduces the plasma cross sectional area. Divertors introduce additional complexities in machines and magnet design, as well as operational risks with the presence of current carrying conductors in regions of high magnetic field.
- X-point configurations, (without conductors within the toroidal magnet cavity) which reduce the plasma cross-sectional area and current carrying capacity for a given toroidal magnet size and capacity. (In Ignitor, X-point configurations with single or double magnetic nulls outside the first wall can be produced for all or part of the discharge if necessary, with relatively little sacrifice in plasma and magnet parameters, i.e., somewhat smaller I_p, and more localized wall loading.)

- Current drive to control q, which may be required to control central sawtooth oscillations at low edge-q_ψ.
- Considerably enhanced confinement regimes above the so-called L-regime.

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Appendix B. The ICRH system

Ion Cyclotron Radio Frequency Heating (ICRH) has been chosen as the most appropriate form of injected heating in view of the high particle densities at which Ignitor is called to operate and taking into account that its technology is currently available. It is desirable that the RF system should be able to deliver at least as much power as that produced by α-particles under relevant ignition conditions (18 - 24 MW). Thus, α-particle heating can be simulated at the same power level in non-reacting plasmas, although α-heating is perfectly axisymmetric and radially more localized than the simulating IC heating.

The frequency range (70 - 140 MHz) has been chosen so that the hydrogen (H) minority heating scheme for operation at the lowest value of the equilibrium magnetic field (5 T) can be adopted, and the ^{3}He-minority heating scheme for other scenarios at higher field, up to 13 T [1, 2].

The complete ICRH system for Ignitor is made of six antennas [3], inserted into recesses provided in the plasma chamber, extending on either side of the horizontal ports. Each of the six antennas has 4 straps forming a 2×2 poloidally and toroidally phased array; each strap is fed by a radiofrequency power generator via a coaxial line and a tuning and matching system. This solution gives the system the maximum flexibility in poloidal and toroidal phasing. Each antenna is protected from the plasma by a Faraday shield. The antennas will have to withstand rather demanding conditions, in particular the periodic heating and cooling cycles (reaching 470 K during baking of the plasma chamber and 30 K in cryogenic conditions) and the stresses induced by major disruptions. Furthermore, their assembly, disassembly and maintenance is to be performed by the in-vessel remote system. All the in-vacuum components can be decontaminated using conventional cleaning procedures; the materials for all the components exposed to tritium contain no mercury, sulfur, chlorine, or other halogens.

A comprehensive analysis of the antenna system performance has been undertaken, including the tuning and matching system. The coupling properties of the antenna have been evaluated employing existing, previously assessed analysis codes, as well as a new, *ad-hoc* self-consistent code developed for this purpose. The analysis was performed assuming 35 kV as the maximum voltage allowed in the whole RF system and an average thermal load of 0.5 MW/m^2 on the Faraday shield. In addition the same reference disruption event scenario envisioned for the entire machine was considered for the design of this shield. According to the existing design, up to 4 MW per port can be injected, corresponding to a total of 24 MW. This design is the result of key contributions from collaborations that are still ongoing with the experts of the Princeton PPL, Oak Ridge NL, and Lodestar Co.

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Appendix C. Diagnostic Systems

A comprehensive set of diagnostics, capable of covering a unique range of plasma parameters (*i.e.*, in B, n, and T) and of investigating alpha particle dynamics and collective modes of different kind, has been studied, taking into account that, in Ignitor, plasma accessibility is provided by six of the 12 large equatorial ports (17×80 cm), and by 12 up-down symmetrical vertical ports (3.5×10 cm).

In particular, CO_2 lasers can be used for simultaneous measurements of densities and magnetic fields, by means of interferometric and Faraday Rotation techniques. To provide an adequate number of chords for meaningful profile inversions be carried out, vertical ports located at 3 different radial positions can be utilized.

Electron Cyclotron Emission (ECE) measurements of T_e can be performed using the plasma emission in the O-mode at the first cyclotron harmonic for relatively low densities ($n_e < 8 \times 10^{20}$ m^{-3}) and at the second harmonic for higher density. The full spectrum range can be covered by a single Michelson interferometer with a resolution time < 5 ms, while higher time resolution can be obtained with a grating polychromator or with a heterodyne radiometer. Measurements of density fluctuations can be performed with microwave reflectometry using the X-mode lower cutoff, or the O-mode cutoff when $T_e < 10$ keV and $n_e < 7.5 \times 10^{20}$ m^{-3}.

Confined alphas could be diagnosed by a Pellet Charge Exchange (PCX) system, requiring pellets with velocities in excess of 2.5 km/sec, which are also required for deep plasma fuelling. Collective Thomson back-scattering of 200 GHz waves in the X-mode can measure the distribution and the spectrum of the confined alphas.

An important role among plasma diagnostics for Ignitor will be played by neutron measurements. In particular, a multicollimator with Magnetic Proton Recoil Detectors is envisaged for measuring the ion temperature profile. Such an instrument, initially proposed and designed for Ignitor [1], has indeed been built and operated at JET [2]. The local neutron emission has been evaluated by the 3-D Monte Carlo code MCNP [3], in which the geometry and nuclear composition of the main components of the Ignitor machine have been included. We note that neutron transport calculations may be usefully employed to study some of the problems encountered in the *in situ* absolute calibration of the neutron detectors, the choice of the most suitable location of these detectors, etc..

References

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