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The Key to ITER: the Divertor and the First Wall

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I. INTRODUCTION

Present world fusion research programmes are directed ultimately towards the construction of a demonstration fusion reactor which should be a full ignition, high power device. These research programmes have merged recently with the signature of the ITER Agreement to conduct jointly the Engineering Design Activities (EDA) of ITER in preparation for construction. The programmatic objective of ITER is to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes.

This turning point in the fusion programmes has been reached as a result of the steady progress of the last decade and the realisation that the challenge in fusion research is to demonstrate sustained ignition in a next step machine. So far, near breakeven conditions have been achieved, with the JET tokamak achieving a fusion triple product, $n_i\tau_E T_i$, of 8-9x10^{20} m^{-3}skeV and the first experiment in deuterium-tritium producing more than one megawatt of fusion power [1]. However, these conditions could only be maintained transiently with MHD instabilities (such as giant sawteeth which induce disruptions, and vertical instabilities) and impurity influxes still limiting the achievement of better performance and steady state operation at high heating power.

The quantitative understanding of fusion plasmas has also improved with the development of the critical electron temperature gradient model for plasma transport [2,3,4]. Good quantitative agreement has been obtained between this model and data from tokamak experiments. The main predictions are also consistent with statistical scaling laws [5] in their domain of validity. Such a model begins to provide a predictive capability for defining the parameters and operating conditions of a demonstration fusion reactor, including impurity levels.

Present experimental results and model predictions suggest that impurity dilution is a major threat to a reactor. The divertor concept for impurity control, plasma fuelling and helium ash exhaust needs to be developed further [6]. The operating domain of a tokamak that uses a particular divertor concept must be demonstrated for both normal and abnormal conditions (eg when MHD events or runaway electrons release energy over a short period of time).

The cooling of plasma facing components in the divertor, and more generally in the first wall and blanket, also raises questions, in particular, the choice of a coolant that complies with the safety regulations. Since the flux of 14MeV neutrons will be about 1-3MWm^{-2} and the power flux at the divertor plate will be about 5MWm^{-2}, an efficient cooling system is essential for the operation of the device.

In this article, these questions are reviewed in the context of ITER and directions for future research and development work and concept development are indicated.

II. THE ITER PLASMA

A. ITER as the Core of a First Reactor

The size and performance of a thermonuclear reactor are largely defined by the constraints of present technology and the extrapolation of present results. The toroidal magnetic field that is technically achievable with superconducting magnets is limited to below 13T on the conductor. Tokamak experiments show that the value of the safety factor at the plasma edge should be above two and that, with a low Z first wall material, energy confinement improves with increasing plasma current and is independent of heating method. These considerations suggest that a thermonuclear reactor should be a large device. If ITER is to demonstrate fusion as an energy source, it should be similar in size and operating conditions to a reactor core [2]. In practice, ITER should sustain full ignition during normal operation, producing fusion power in the range of 1GW for about 1000s. In addition, the basic configuration, aspect ratio and elongation of ITER should be similar to those of operating tokamaks, such as JET and DIII-D.

B. A Plasma Model for ITER

To define the operating conditions of ITER, a fully coherent, time-dependent transport model has to be developed. For example, the approach to ignition and the control of ignition (burn control) and burn products (helium "ash") requires a model which addresses all aspects of energy and particle transport, together with a description of "sawteeth", β-limit instabilities and the edge plasma (the separatrix, the scrape-off layer and the divertor). One such model, which is consistent with available experimental results and statistical scaling laws (such as ITER89-P [5] for L-mode energy confinement) in their domain of validity, is the **Critical Electron Temperature Gradient** model of energy and particle transport [2,3,4]. In this model, electrons determine the degree of confinement degradation; ion anomalous transport with heat diffusivity χ_i linked to electron heat diffusivity χ_e; anomalous particle diffusivities, D, for ions and electrons, proportional to χ; and an anomalous particle "pinch", V, related to the profile of the safety factor, q [4].

Specifically, above a critical threshold, $(\nabla T_e)_c$, in the electron temperature gradient, the transport is anomalous and greater than the underlying neoclassical transport. The electrons are primarily responsible for the anomalous transport, but ion heat and particle transport are also anomalous. The general expressions for the anomalous conductive heat fluxes are:

$$Q_e \equiv -n_e \chi_e \nabla T_e = -n_e \chi_{an,e} (\nabla T_e - (\nabla T_e)_c) H(\nabla q)$$

$$Q_i \equiv -n_i \chi_i \nabla T_i$$

$$\chi_i = 2\,\chi_e \sqrt{T_e/T_i} \times \{Z_i/\sqrt{(1+Z_{eff})}\}$$

The anomalous coefficients for particle transport are:

$$D = 0.5\,\chi$$

$$V = -D\,(\nabla q)/2q$$

The critical electron temperature gradient model [2,3,4] specifies possible dependences for $\chi_{an,e}$ and $(\nabla T_e)_c$. To complete the plasma model requires a description of sawteeth, β-limit instabilities and the edge plasma (the separatrix, scrape-off layer and divertor) for which rudimentary models are also included.

The model exhibits the following features of experiments: consistency with physics constraints, global scaling laws and statistical analyses; a limitation in the electron temperature; no intrinsic degradation of ion confinement with ion heating power; no dependence of confinement on mass; and similar behaviour of particle and heat transport. The model, which has no free parameters, reproduces plasma profiles for a wide variety of discharges in ohmic, L-mode and H-mode (high confinement) regimes in various tokamaks. In particular, the simulation of off-axis heating [7] is almost a direct confirmation of a $(\nabla T_e)_c$, while electron heat pulse propagation studies [8] show a diffusivity, $\chi_{HP} \sim \chi_{an,e} > \chi_e$. The existence of the hot-ion mode is consistent with a critical gradient associated with the electron temperature and current ramp experiments are consistent with the effects of magnetic shear modifying the dependence of confinement on poloidal magnetic field. It should be noted that in the plasma interior, the same model applies to the L- and H-regimes and particle and energy confinement improve together.

C. A First Reactor

The parameters of a **first reactor** are defined by technology and physics predictions. The minor radius of the reactor plasma needs to be about twice the thickness of the tritium breeding blanket, which makes it approximately 3m. A practical aspect ratio of between 2.5 and 3 sets the plasma major radius to 8 or 9m. The elongation of the plasma must be limited to a value less than two. Safe operation can be assumed for a cylindrical safety factor greater than 1.6. Plasma physics requirements can be fulfilled by operating at a toroidal magnetic field of about 6T. This defines a reactor with a current capability of about 30MA. The total magnetic flux available could be about 1000Wb. The reactor will operate with a D-T mixture, and helium "ash". Impurity control will be achieved by plasma flows in an appropriate divertor configuration. Sawteeth will be beneficial in ejecting helium from the central plasma. The reactor plasma will most likely be characterised by a temperature of 25keV and a density greater than $10^{20}m^{-3}$.

D. Modelling ITER

In ITER, the duration of the discharge in normal operation need not exceed 1000s. In comparison with a **first reactor**, the inductive flux can therefore be lower and the major radius somewhat smaller. Nonetheless, the operating conditions in ITER should be comparable to those of a first reactor and ITER should be considered as the core of such a reactor.

Typical configurations for JET and a **reactor core** are shown in Fig.1(a). The full energy and particle transport model is solved in the simulation geometry (major radius, R=7.75m, horizontal minor radius, a=2.8m, toroidal magnetic field at R, B_T=6T, plasma current, I=25MA, and plasma elongation, κ=1.6) shown in Fig. 1(b). A D-T fuel mixture is assumed and the transport of helium ash, created during the D-T fusion process, is modelled. 1% of the total recirculation (helium and D-T fuel) in the divertor is pumped. Impurity control is assumed to limit the concentration of beryllium impurities to 1% of ions.

Simulations using the model described in Section II.B show that the reactor core operates well in L-mode and at high power. Low power operation is possible in a clean plasma, but high helium concentrations (helium "poisoning") precludes such operation. Contrary to the Goldston scaling law [9], which suggests that the fusion triple product, $n_i\tau_E T_i$ is constant when τ_E degrades as $P^{-1/2}$, the degradation of confinement with the transport model of Section II.B saturates at high power, P. Thereafter, $n_i\tau_E T_i$ increases with power until the β-limit [10] is reached.

In these simulations, ignition is achieved with 10MW of ion cyclotron resonance heating. Burn control at various power levels is achieved with fuel injection controlled by feedback on the power produced. The operating density is then determined by the conservation equations for energy and density. With a feedback system, ignition can be sustained for a wide range of powers (Fig. 2) above a minimum α-power, P_α of approximately 0.2GW. The corresponding minimum density is high (about $10^{20}m^{-3}$) and is compatible with impurity control concepts foreseen at present to rely on energy removal by neutrals and radiation in a divertor (Section III and reference [6]). Higher power ignition is achieved at even higher density and stored energy but, generally, at lower temperature. The confinement time decreases from 4s when P_α=216MW to 2s when P_α=850MW and the global Troyon factor [10] increases from 1.4 at 216MW to 3 at 850MW (Table I). In all cases the density profile is slightly peaked (Fig. 3) with edge fuelling being sufficient to fuel the centre. Steady ignition conditions are achieved with a relatively high helium concentration (~20-25%): without sufficiently high transport and adequate pumping, helium poisoning can quench the ignition. In fact, while the H-mode might have short term benefits for approaching ignition, long term deficiencies due to helium poisoning can arise (Fig.4). Furthermore, the compatibility of the H-mode and high density operation has yet to be established.

For the range of conditions considered, the bootstrap current increases from 2.7MA at 216MW to 7.1MA at 850MW and for relatively flat density profiles the bootstrap current tends to occur near the plasma edge. Furthermore, the loop voltage is similar in all cases (~0.1V), the resistive flux consumption is quite low and a magnetic flux of 360Wb is sufficient to provide one hour current flat-top operation. Increasing the radius of the central solenoid of the reactor core by 0.8m would make available a further 360Wb and provide an extra hour of steady operation. At present, a continuously operating reactor with non-inductive current drive would require [11] a convincing demonstration of a current drive efficiency exceeding $1x10^{20}A/m^2W$

for a density above $10^{20}m^{-3}$ or high power operation in a regime with a dominant bootstrap current. Otherwise, ignition and adequate impurity control must be demonstrated at the lower density of $5x10^{19}m^{-3}$ (when a current drive efficiency of $0.5x10^{20}A/m^2W$ would suffice) but this would aggravate the divertor problem.

III. THE DIVERTOR

A major challenge in fusion research is the achievement of high power exhaust, impurity control, high helium pumping and the rapid re-circulation of several grammes of tritium per second. At present, the divertor appears to offer the greatest potential for meeting this challenge. The divertor configuration, with an X-point inside the vacuum vessel, channels particle and energy flows along the open magnetic field lines just outside the separatrix towards a localised remote target and pumping region. The principal source of impurities is thus well-removed from the main plasma, but sputtered impurities cannot be eliminated completely and these have then to be retained in the divertor region against their natural tendency (due to the thermal force [12] acting on impurity ions) to contaminate the plasma core.

However, a fully coherent and robust divertor does not exist and is one of the most difficult challenges of the ITER EDA. The heat load on the targets of a conventional divertor would be concentrated in a scrape-off layer (SOL) thickness of between 5mm and 10mm and this thickness does not increase with the size of the tokamak and tends to decrease with improving confinement. Such a concentrated heat load would lead to excessive impurity production and plasma contamination and raises questions about the effect of erosion, sublimation, melting and redeposition. Furthermore, the thickness of target plate tiles will have to be limited to a few mm in order to ensure adequate heat transfer and this raises questions about the vulnerability and associated safety aspects of an actively-cooled system from which the coolant could leak.

The impact of impurities from plasma facing components can be minimised by (a) selecting a material of low atomic number which would be re-deposited mainly in the divertor region and (b) eliminating the direct interaction of the plasma with the target. Beryllium seems to be one of the best choices of material in view of its good thermal-mechanical properties and the possibility of low tritium retention. An intensive research and development programme will be needed to qualify beryllium as a target material.

Nonetheless, for a reactor producing a fusion power in the range 3-6GW, the heat load on the targets would be in the region of $100MWm^{-2}$, or more, even when optimistic assumptions are made for the inclination of the magnetic field to the target plate and the expansion of the magnetic flux. This heat load would be too high for reliable operation and methods proposed so far to alleviate this problem (eg. "sweeping" the plasma over the divertor target plates) might even aggravate the problem because thermal fatigue would result from the increased and faster thermal cycling of the targets. A new divertor concept is required and several possibilities must be developed and tested.

A. An Advanced High Density Divertor

A solution for an advanced high density divertor is to exhaust energy by energetic neutrals (created by ion-neutral collisions, such as charge-exchange) and radiation and to extinguish the SOL plasma before the material target is reached [6]. If a sufficiently large flow of low energy neutrals (somewhat above room temperature) is injected perpendicular to the plasma flow in the divertor region, the parameters of the SOL plasma will be constant along lines inclined at an angle, α to the magnetic field. If the divertor is sufficiently large and dense, the layer will be opaque to the incident cold neutrals which will not cross the SOL. If the plasma temperature reaches a few eV at the target plates, the plasma flow will become supersonic near an ionisation front, inclined at an angle, α, and the plasma "flame" will be extinguished before the target plates are reached [6]. In reality, the angle α will be a function of distance along the magnetic field. This structure is confirmed by 2-D calculations [13]. However, when the SOL is opaque to cold neutrals, the essentials of the problem can be determined by a 1-D code which must, nonetheless, include impurity retention and radiation, ion-neutral and electron-ion collisions and recombination and re-ionisation outside the SOL.

B. Geometrical Condition to Extinguish the SOL Plasma

The geometrical condition to extinguish the SOL plasma before the target plates are reached is seen most readily as a condition on the angle, α' in the projection of the magnetic field in the poloidal plane (Fig. 5). If ℓ is the projected length of the magnetic field and Δ is the SOL thickness, the SOL plasma is extinguished before the target plates are reached if:

$$\mathrm{Tan}\alpha' > \Delta/\ell$$

where $\alpha'=\alpha B_T/B_P$ relates the angles, α' and α, in the poloidal projection and along the magnetic field. α will comprise two parts if neutrals are incident from both the private flux zone and from the external region of the SOL.

C. Physics Condition to Extinguish the SOL Plasma

A physics condition for extinguishing the SOL plasma before the target plates are reached can be obtained from the above geometrical condition, together with the solutions to the conservation laws for particles, momentum and energy for one-dimensional flow along the open magnetic field lines in the SOL. It is found [6] that the condition to extinguish the SOL plasma can be written:

$$2 \times 2\pi R \ell n_o V_o E_H > (P_{in} - P_{rad})$$

That is, the power transmitted by a particle flux, n_oV_o at an energy, E_H across two surfaces, each of area $2\pi R\ell$, must exceed the net input power, where P_{in} is the input power and P_{rad} is the radiated power. This condition shows that the product $n_0\ell$ is independent of the SOL thickness and is proportional to the net input power, for fixed E_H. Furthermore, n_0 and ℓ cannot be separated.

The structure imposed on the divertor channel by the cold neutral flow allows kinetic energy in the hot neutrals and radiation to be distributed more evenly over the whole of the divertor channel. The remaining power will be transferred to the targets by charged particles, in a narrow SOL thickness with the ions being subjected to sheath acceleration prior to sputtering at the target. It should be pointed out that the energy of the hot neutrals will also be sufficiently high for sputtering to occur and that this might present problems for targets made from high-Z material. The geometry must therefore accommodate the hot neutrals, organise the cold neutral flow and capture the sputtered impurities.

Under the divertor conditions proposed, plasma flows are generated by the ionisation of the incident cold neutral flux. In principle, a strong flow of deuterium, which becomes supersonic across the shock wave and which is directed towards the targets, can prevent the back diffusion of impurities with frictional forces overcoming thermal forces [14]. At the same time, the neutral density should be high in the private flux zone of the divertor and the helium pumping requirement is reasonable. However, a feedback system to control the neutral density might be needed and account should be taken of sputtered material and helium radiation in the cold plasma.

E. A Possible Implementation of this Advanced Divertor Concept

A possible implementation of this advanced divertor concept is shown in Fig. 6. Two separate areas receive the power load associated with normal operation and abnormal events, such as disruptions (see Section IV.A.). During normal operation the power load is reduced and exhausted by energetic neutrals over a large area along the divertor channel. The power exhaust perpendicular to the magnetic field would be maintained well below $5MWm^{-2}$. Bumper targets would receive substantial loads only as the result of abnormal events.

For typical conditions foreseen for the midplane of the SOL in ITER, namely a plasma pressure, p_∞=2.7x10^4P (corresponding to T_i=1500eV, T_e=750eV and n=7.5x$10^{19}m^{-3}$) and q_∞=1990MWm^{-2} (corresponding to a total input power, P_{in}=400MW, a major radius, R=7.75m, a SOL thickness, Δ=0.01m and a ratio of toroidal to poloidal magnetic field, B_T/B_P=5), the SOL plasma can be extinguished in a divertor of length, ℓ=2.5m. It is assumed that β=0 and that the total radiated power, P_{rad}=0. The solutions at the targets are:

$$V_y = -1.47\text{x}10^5\text{ms}^{-1};\quad n = 3.7\text{x}10^{20}\text{m}^{-3};\quad n_oV_o/Tan\alpha = -5.48\text{x}10^{25}\text{m}^{-2}\text{s}^{-1};\quad E_H = 226\text{eV}$$

$$\alpha = 8\text{x}10^{-4} \quad\Rightarrow\quad n_o = 1.46\text{x}10^{19}\text{m}^{-3} \quad \text{for} \quad V_o = 3\text{kms}^{-1}$$

The power exhaust perpendicular to the magnetic field would be maintained below $5MWm^{-2}$, taking account of a peaking factor. An energetic plasma in the SOL is thus converted into a cold plasma and neutrals at a location in the divertor channel that depends on the power load. The concept leads to a divertor geometry that requires about a quarter of the volume available to the plasma and is quite different from that considered until now for ITER. The concept needs to be tested on present tokamaks.

IV. THE FIRST WALL

It is necessary to accommodate the power produced not only during normal operation but also during abnormal operation, when MHD events (such as disruptions, giant ELMs or giant sawteeth) and runaway electrons release energy in such a short time that the SOL "flame" extends and is no longer extinguished within the divertor. Steps must be taken to ensure that these abnormal events rarely occur and that their impact is minimised by the introduction of sacrificial elements.

A. Avoiding and Safeguarding Against the Effect of Abnormal Events

Operation must be restricted to within known limits. Density limit disruptions may result from an unfavourable power balance, involving atomic processes (such as radiation and charge-exchange) near the q=2 surface. A reactor should operate far from the density limit since the α-power should far exceed impurity radiation losses near the q=2 surface and low Z material should be fully ionised outside the q=2 region. Beryllium, which leads to redeposition as powder (rather than the co-deposition of carbon), could be used as a first wall material and the risk of density limit disruptions eliminated. Disruptions can also result from "locked" modes which occur on transition to the X-point configuration, with rational q surfaces at the edge being coupled to resonant perturbations of the magnetic field. Error fields, which result from the slight misalignment of coils, could be corrected by a set of multipolar coils placed outside the vessel. Additional active feedback control of potentially unstable modes could be effected by the use of internal saddle coils. The vertical stability of the plasma can also be assured by limiting the elongation to a value of about 1.6.

Since disruptions might not be avoided completely, it will be necessary also to introduce safeguards that will reduce their impact. The resistance of the vacuum vessel should be low, so as to limit the dynamical and mechanical effects of disruptions outside the vessel. Sacrificial elements, such as bumpers and limiters, should also be introduced to take the brunt of disruptions, ELMs and runaways, and prevent major structural damage. In addition, the interaction of an instability with the first wall and divertor may lead to impurity material that falls through the plasma, producing a disruption. To minimise the occurrence of this event, the divertor should be located at the bottom of the vessel.

A.1. Disruptions

In a disruption, the magnetic field reconnects, the plasma energy is redistributed and is exhausted through the SOL on a short time-scale (~10ms). It is unlikely that low Z material could radiate all this power and bumpers will therefore be needed to radiate part of the power and to accommodate the energy by melting and vapourizing. The vapour, however, will be re-deposited over the interior surfaces of the tokamak. Since the first layer of the inner wall must be refreshed periodically by, for example, the redeposition of evaporated beryllium, it would appear desirable to maintain a low-Z compatibility and use bumpers of solid beryllium blocks contained in a box and in thermal contact by a film, or bath, of gallium. The blocks could then be exchanged without cutting the cooling channels.

A.2. The L-H transition and ELMs

In the absence of atomic processes in the edge plasma, the H-mode may be the natural consequence of the transport model of Section II, since $\chi_{an,e}$ depends on shear, and reduces towards zero near the separatrix, leaving a narrow edge layer of neoclassical ion transport. The transition from the H-mode to the L-mode is then related to the rapid destruction of this confinement barrier and the subsequent large increase in anomalous transport in the edge. In fact, an ELM could be thought as the rapid transition between an H-mode and an L-mode. ELMs can also be associated with giant sawteeth and other MHD phenomena such as those that limit most of the high power, high performance H-mode discharges on JET, including those with tritium. The result is a very large increase in the power exhausted into the SOL, the subsequent burn-through of the plasma "flame" to the bumpers and the excessive production of impurities due to beryllium melting or a carbon bloom. The power associated with these transient ELMs could only be tolerated when the peak power deposited remained comparable to the mean power deposited.

A.3. Runaway electrons

Runaway electrons are produced in low density, low temperature plasmas with a high Z_{eff}, where the resistive electric field is large and above a critical value. Normally, these conditions should not occur with a beryllium first wall, but it would be expedient to provide protection against runaway electrons and direct plasma contact (particularly at the start of the discharge) by introducing a series of poloidal limiters (made from carbon or beryllium blocks, cooled, and using a technology similar to that envisaged for the bumpers in the divertor).

B. Tritium Inventory

Tritium will be found inside the vacuum chamber, residing in the first wall due to diffusion (metal, carbon, porous elements), chemical bonding (co-deposition with carbon), exchange with chemical hydrogen already linked like residual water and tritium surface bonding (up to 1kg of tritium could be found in carbon powder that would arise from the erosion of limiters). A relatively small total tritium inventory of a few hundred grammes seems to be possible with a ceramic type blanket. Together with helium, about 1kg of tritium will have to be pumped per hour. The deuterium-tritium mixture can be separated from helium by cryo-condensation or a membrane. Helium can be pumped through turbopumps or cryo-absorption with active carbon or by diffusion. Tritiated impurities (such as methane and water) will also have to be separated. Within the cryostat volume, a total tritium inventory of about 1kg could be achieved, but this would require the tight control of materials, well-chosen blanket elements and pumping system with the capability of recirculating tritium, by, for example, reinjection using centrifugal pellet injection (or by capillary action).

C. Coolant

It is possible that the ITER divertor will have to handle 1GW of α-power (a power flux of over $10MWm^{-2}$) and the first wall and blanket will have to handle 4GW of neutron power (a power flux of over $2MWm^{-2}$). It will therefore be necessary to cool the divertor, first wall and blanket. The coolant will have to be properly selected and will impact strongly on the concept of the reactor and its pipe-work.

The first wall and blanket, considered as an integral entity, will operate in hot conditions in order to ensure adequate cleanliness and outgassing. The temperature at its surface would need to be maintained above 200°C, and preferably above 300°C. The coolant would need to comply with safety regulations and this raises questions about the use of water due to the required high pressure in the divertor region, the contamination of water by tritium permeation through metallic pipes and the galvanic corrosion of these pipes. There could also be a risk of explosion if an air intake occurred and reacted with hydrogen. If, however, the vacuum vessel and the blanket are located inside the vacuum of the cryostat, the risk of an air ingress may be alleviated. For these reasons, water does not appear to be a good candidate for cooling the divertor and the blanket.

A gaseous or liquid coolant, such as helium, may be considered but more space might then be needed in the blanket and the necessarily high pressure and velocity would require the use of a pressurized vessel. Additional space would also be required for neutron shielding. The thermal capacity of the coolant could be increased by loading it with granules to produce a multiphase solid-gas coolant, but such a system would have to be developed.

Another possibility would be the use of a liquid metal, such as gallium, sodium, NaK, lithium or even lead, but the magnetohydrodynamic forces which would arise when the coolant flows across magnetic flux may require the introduction of semi-insulated structural pipes. A liquid metal coolant could operate at low pressure and could be self-healing in the event of a crack developing.

Another original solution is based on cooling the blanket by internal thermal convection, with the heat being extracted by conduction to the vacuum vessel.

The question of cooling the divertor system has yet to be resolved and depends largely on the divertor concept adopted.

V. AN INTEGRATED DESIGN FOR A NEXT STEP TOKAMAK

Of course, all these different reactor issues have to be integrated into a coherent and cost-effective design which eases the problem of stress, limits the cost and ensures the reliability of ITER. Its construction must be based on what is known and it must achieve high levels of simplicity, reliability and safety, and yet provide flexibility. It should: demonstrate sustained high power, semi-continuous operation (eg. more than 1GW for 1000s); study the operating conditions of a reactor; provide a testbed for the study and validation of tritium breeding blanket modules in reactor conditions; test the first wall technology; and define the exhaust and fuelling requirements. Furthermore, the overall capital cost must be tightly controlled and the ratio of this cost to the thermal power output must be in the range relevant to other sources of energy. Basically, ITER must be the core of a reactor if it is to demonstrate fusion as an energy source.

A possible configuration which would achieve these objectives is a tokamak with a plasma current of up to 25MA, a toroidal magnetic field of 6T, a major radius of about 7.5m, a minor radius of about 3m, and an elongation of 1.6 (Fig.7). Energy exhaust and impurity control are addressed by high density operation and a pumped divertor. The approach to ignition could utilise low power ion cyclotron resonance heating, while long pulse ignition (~1/2 hour) would be sustained with possibly X-point, L-mode confinement at a power of several GW. With sustained ignition conditions, blanket modules could be tested under neutron fluxes of over $1 MWm^{-2}$.

VI. CONCLUSIONS

Recent results and model predictions allow the size and operating conditions for a fusion reactor to be predicted with some confidence. Simulations show that the reactor core of Section II would operate in L-mode and at high power. The advantage of better H-mode energy confinement is offset by increased levels of helium "ash". Ignition can be maintained for a wide range of α-powers, but above a minimum, approximately equal to 0.2GW. The corresponding minimum plasma density is about $10^{20} m^{-3}$ and central plasma temperatures are about 25keV.

In the period leading up to the construction of such a reactor core certain critical issues must be resolved. A fully coherent and robust divertor for power exhaust and impurity control is one of the most difficult challenges that will be encountered during the ITER EDA. For a reactor core producing thermal power in the range of several GW, the heat load on the targets of a conventional divertor would be too high for reliable operation. Further development of the divertor concept is required and several possiblities must be tested. An acceptable solution is most likely to be achieved with a high density divertor that utilises: energy loss by charge-exchanged neutrals; a radiative or cold plasma or gas target; and high plasma flows for impurity retention.

It is also necessary to accommodate the power exhausted during abnormal operation, such as arises from disruptions. Operating restrictions, control coils and sacrificial elements will be necessary to help avoid disruptions and limit their impact.

The choice of cooling material has a strong impact on the technology for the advanced materials and components needed for the first wall and blanket of the reactor and on the design of these components. Liquid metal coolant may offer a practical solution which could be extended to a breeding blanket design if, in particular, the possibility of a bi-phased liquid metal with granules is considered. However, the need to maintain an insulating layer to ensure the flow of coolant and to prevent the corrosive effect of some liquid metals must be studied further. The possibility of using low pressure water in the first wall blanket and divertor structures may be precluded for safety reasons.

High quality components with sufficient reliability will need to be manufactured. Industry must be involved at an early stage to enable fabrication on a production basis.

The final challenge is the integration of all these different reactor issues into a coherent and cost-effective design which eases the problem of stress, limits the cost and ensures the reliability of a Next Step tokamak which is, in fact, a reactor core.

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Table I

Parameter	Case 1	Case 2	Case 3	Case 4
P_α(MW)	216	427	637	850
$T_i(0)$(keV)	19	23	21	16
$<n_e>(10^{19}m^{-3})$	11	16	21	26
τ_E(s)	4.0	3.4	2.8	2.1
I_{boot}(MA)	2.7	4.8	6.6	7.1
n_{He}/n_e (%)	19	24	24	20
V_{loop}(V)	0.15	0.10	0.10	0.11
g_{Troyon}	1.40	2.34	2.86	2.97

These values were obtained using the full energy and particle transport model for electrons, D-T ions, helium and a specified concentration of beryllium in an X-point, L-mode tokamak configuration.

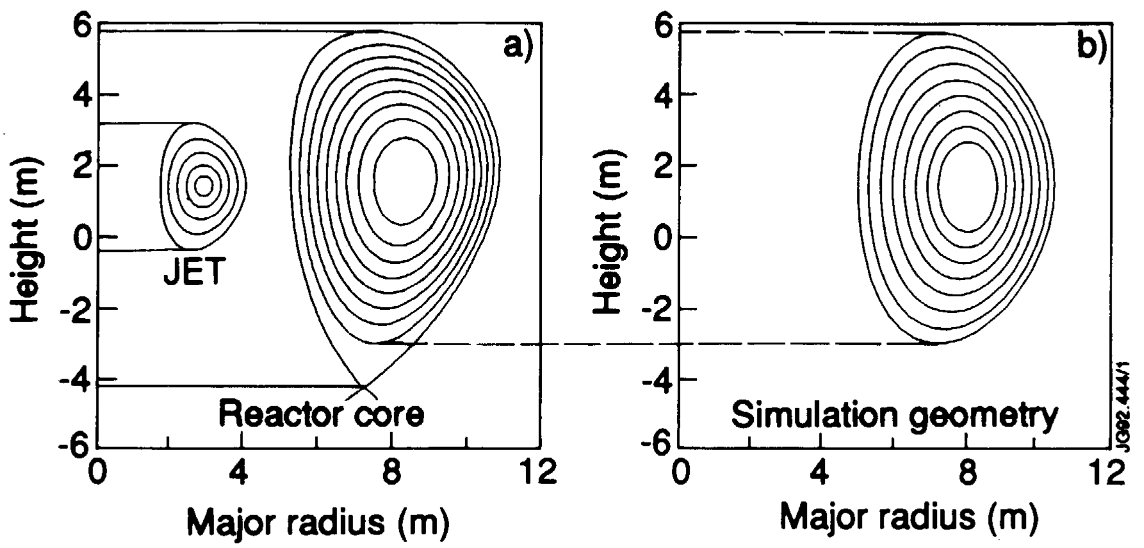

Fig. 1: (a) Typical configurations for JET and a reactor core and (b) the simulation geometry used to solve the full energy and particle transport model.

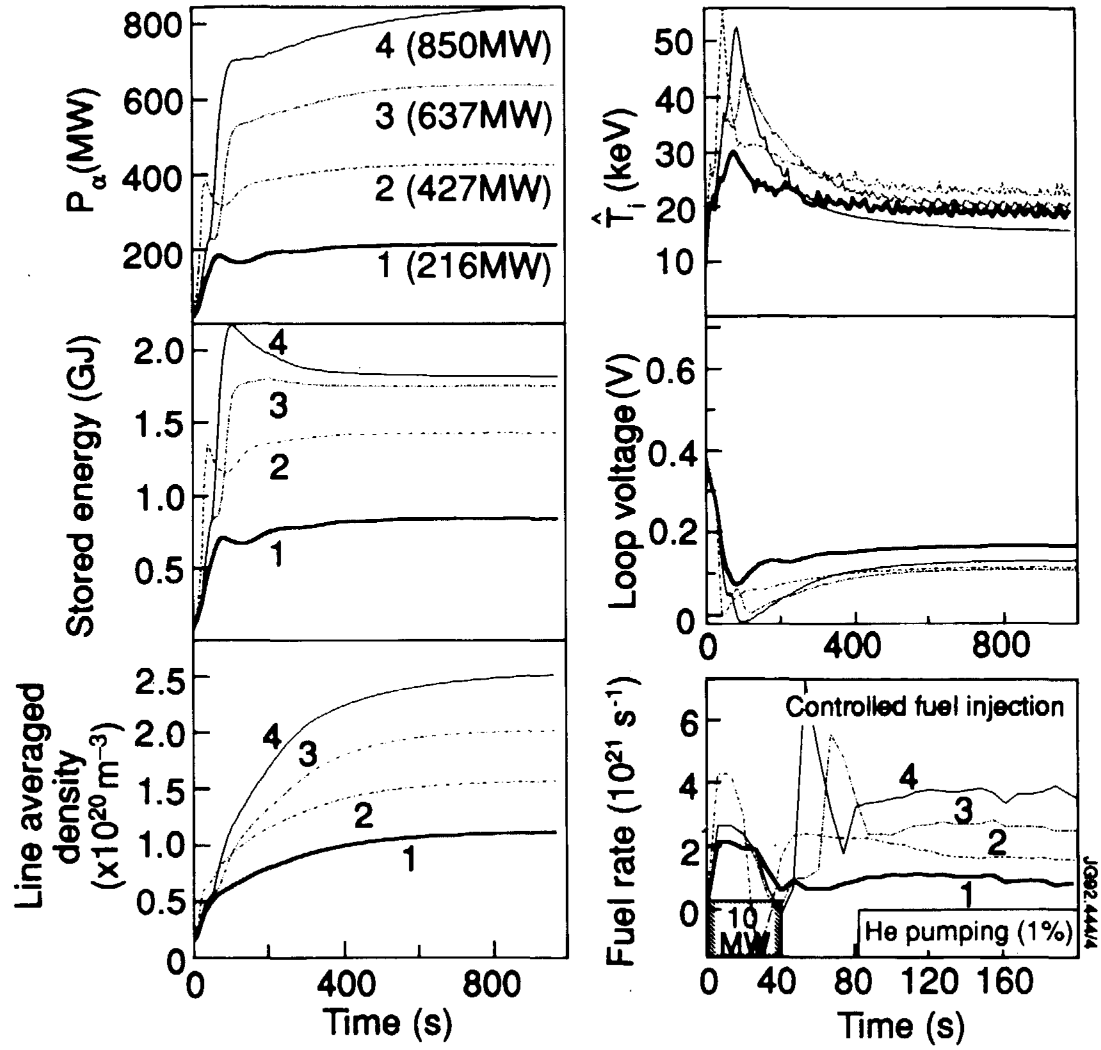

Fig. 2: Simulations of the start-up and burn control of a reactor core at various power levels. Shown are the temporal evolution of α-power, stored energy, line-averaged density, central ion temperature and loop voltage. The RF heating needed for ignition, the fuel injection (controlled by feedback on the power produced) and the helium pumping are also shown.

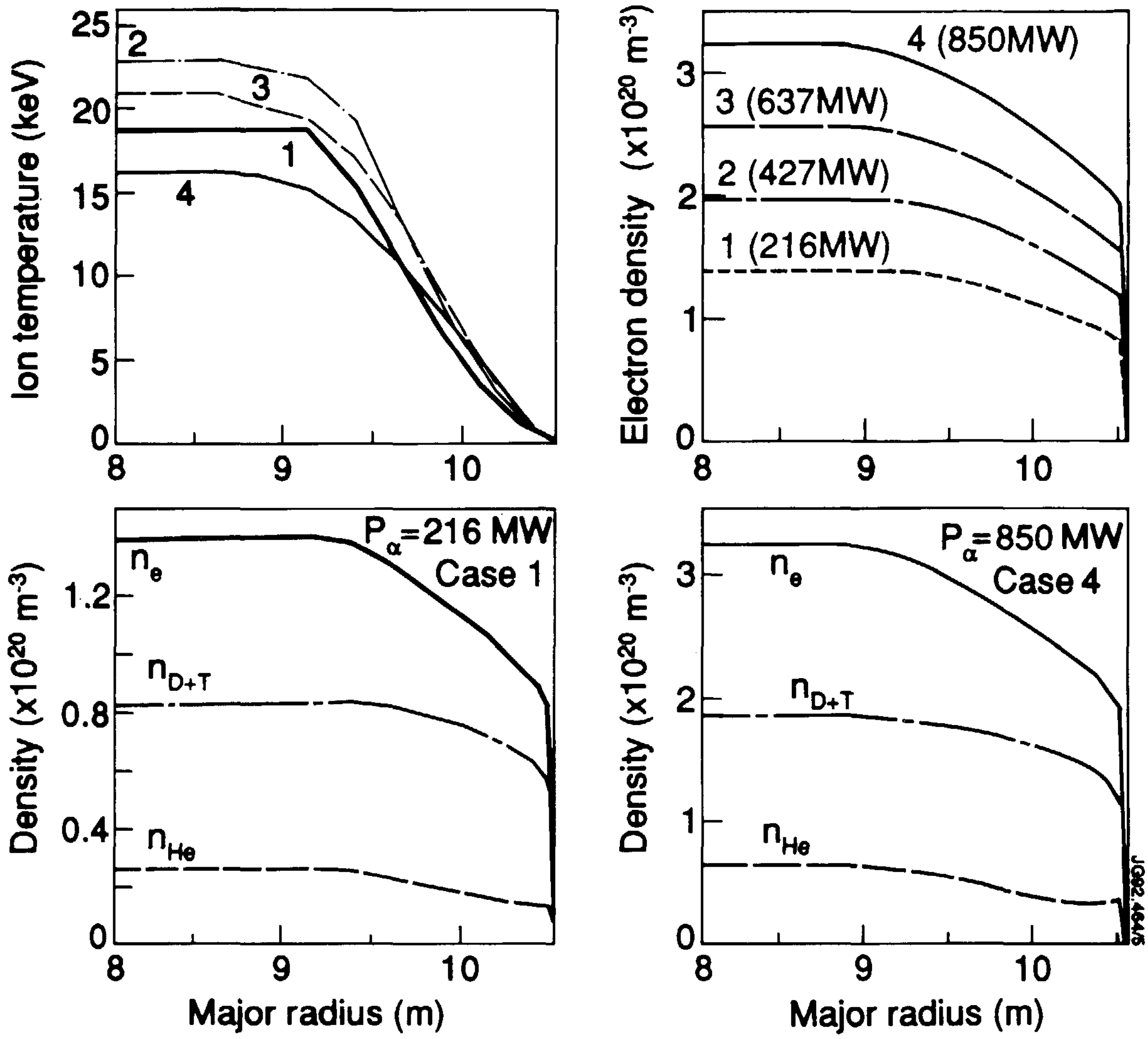

Fig. 3: Steady-state profiles of ion temperature and electron density calculated for a reactor core at various power levels. Shown also are the density profiles at nominal α-powers of 0.2GW (Case 1) and 0.8GW (Case4).

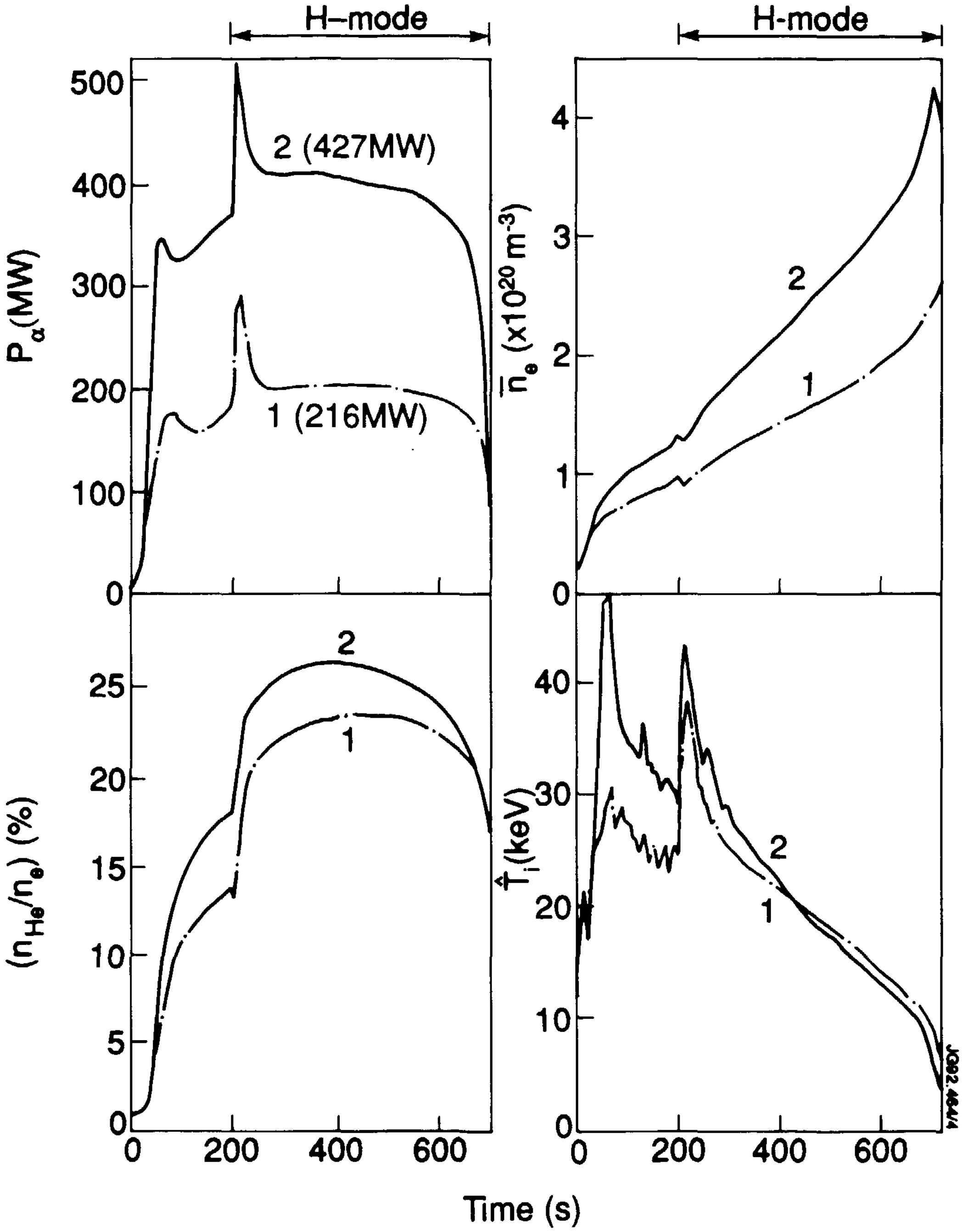

Fig. 4: The long term deficiencies of helium poisoning are shown by modelling the H-mode during the ignited phase of a reactor core. The nominal α-powers are approximately 0.2GW (Case1) and 0.4GW (Case2).

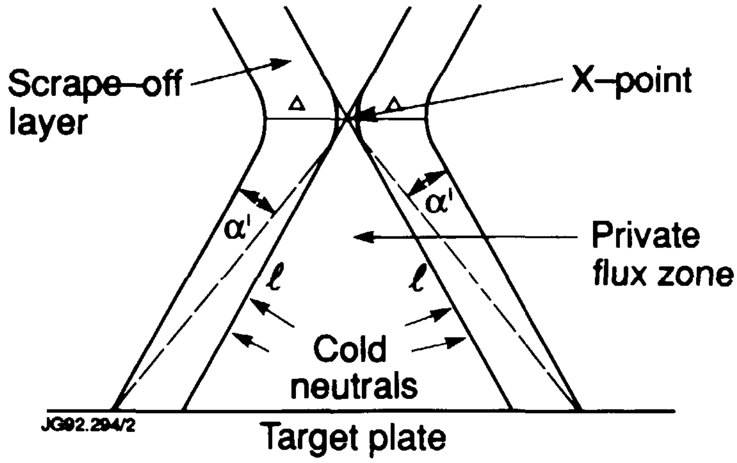

Fig. 5: Schematic of poloidal projection of the scrape-off layer plasma.

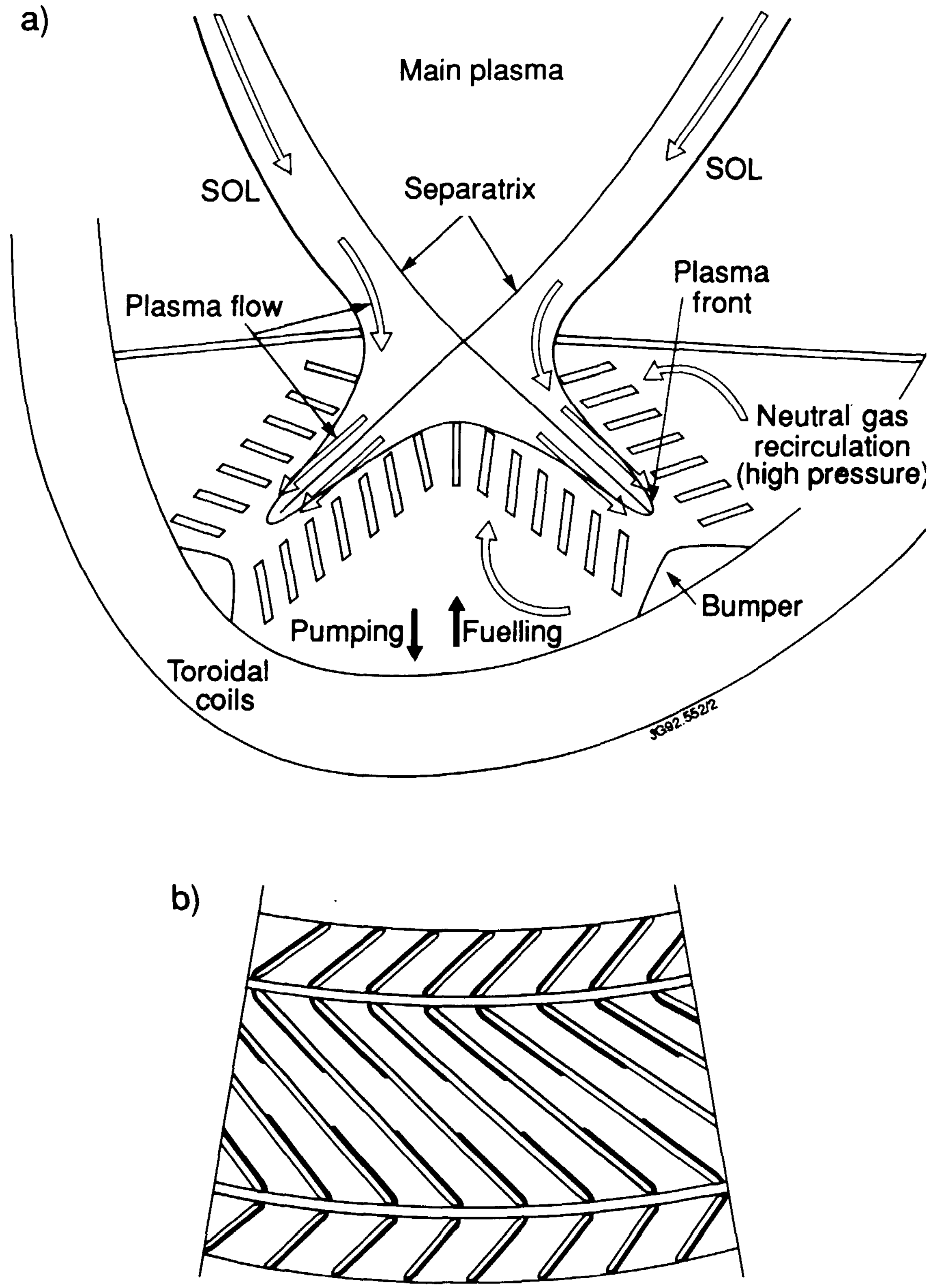

Fig. 6: An advanced divertor concept shown schematically in (a) poloidal cross-section and (b) plan view.

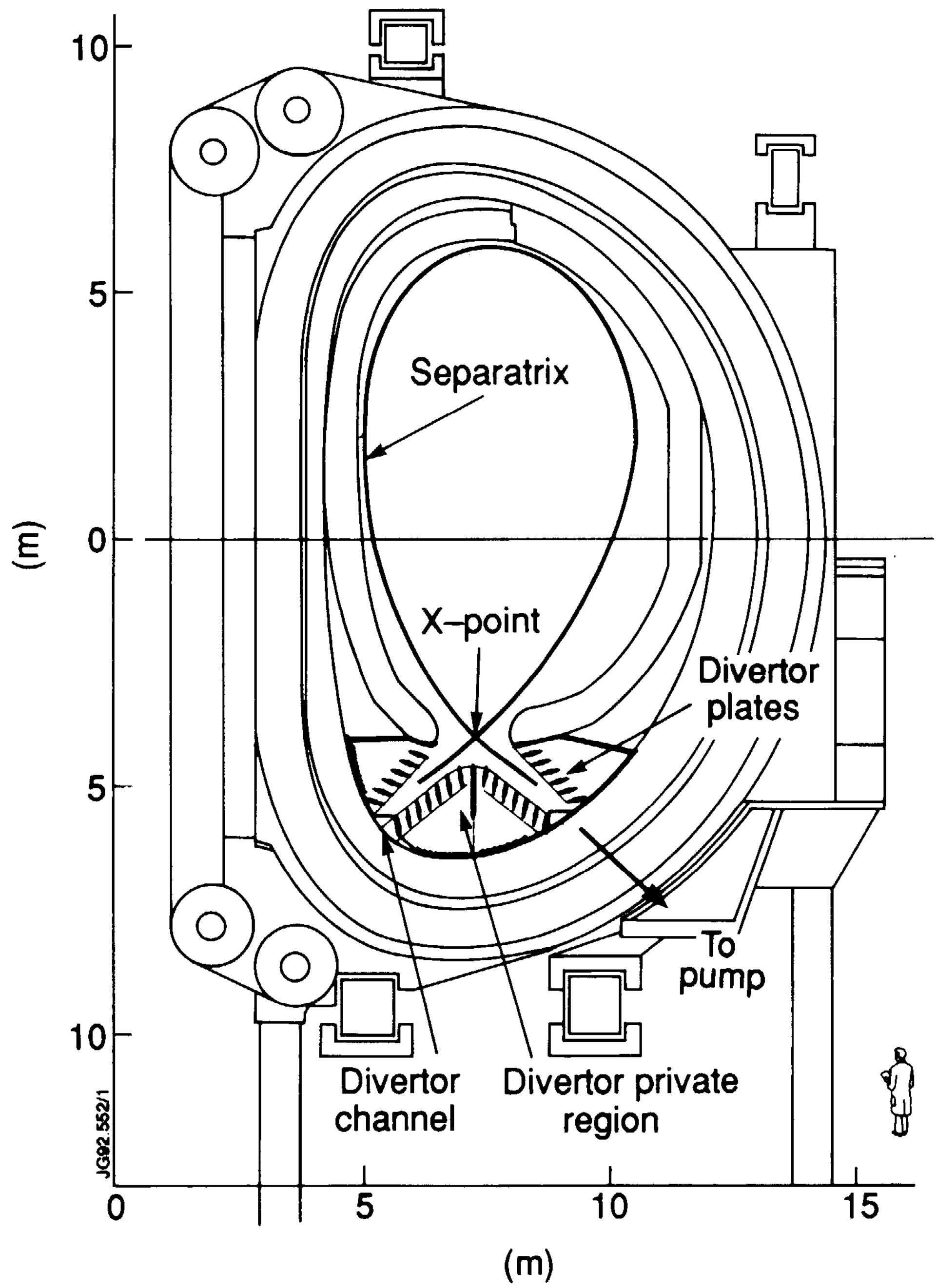

Fig. 7: A possible Next Step configuration.