FUSION MACHINES

M. Van Schoor and R. R. Weynants
Laboratory for Plasmaphysics,
Association 'Euratom-Belgian state',
Ecole Royale Militaire - Koninklijke Militaire School,
B-1000 Brussels, Belgium

ABSTRACT

A concise overview is given of the principles of inertial and magnetic fusion, with an emphasis on the latter in view of the aim of this summer school. The basis of magnetic confinement in mirror and toroidal geometry is discussed and applied to the tokamak concept. A brief discussion of the reactor prospects of this configuration identifies which future developments are crucial and where alternative concepts might help in optimising the reactor design. The text also aims at introducing the main concepts encountered in tokamak research that will be studied and used in the subsequent lectures.

I. INTRODUCTION

Very soon after the discovery of nuclear fission in 1938, the possible peaceful application of this new source of energy was recognised and commercial power plants became available. Stimulated by this succes, first concepts for the peaceful use of fusion energy emerged well over 50 years ago. In his opening speech to the first Conference on the Peaceful Uses of Atomic Energy held in Geneve in 1955, H.J. Bhabha ventured to predict that " *a method will be found for the liberating fusion energy in a controlled manner within the next two decades*". Nevertheless, some people at least were aware of the severe problems that would have to be solved. Indeed, in the first article on the fusion issue published in 1956, R.F. Post wrote; " *However, the technical problems to be solved seem great indeed. When made aware of these, some physicists would not hesitate to pronounce the problem impossible of solution*".

Dispite the latter statement, but well aware of it, a world wide R & D campaign was launched to devellop a nuclear fusion reactor. Surprisingly the basic concepts, which nowadays are considered to be the most successful and promessing, had already been published at that time, albeit offcourse without all the plasma physics knowledge available today and without the techniques and insights needed for a proper scale demonstration[1].

Two main lines are pursued towards the realisation of thermonuclear fusion: inertial (ICF) and magnetic confinement (MCF). In both cases, a burn criterion must be satisfied which requires that a minimum quantity of fuel, represented by the fuel density n, be maintained together for a minimum time span τ_E (the energy confinement time) at a sufficiently high temperature T, brought together in the fusion triple product $n\tau_E T$. Both of these lines have achieved considerable progress in recent years and in both instances the prospects for successful reactor application have been strengthened. In this lecture the basic principles of each of these lines are given, followed by a more in depth discussion of the configurations in which magnetic fusion research is pursued, with special emphasis on the tokamak

II. INERTIAL CONFINEMENT

Inertial confinement fusion[2] (ICF) uses laser or particle beams (called drivers) to heat frozen D-T pellets (radius R), either directly or indirectly via conversion into X-rays, to the necessary fusion temperatures[2] . The heating pulses are typically 1 to 10 ns long. A reactor based on this concept is inherently pulsed and, hence, the basic reactor requirement should be to produce a substantial target gain G, defined as the energy yield of the fusion reactions divided by the energy of the driver. High yield depends on the number of fusion reactions that can occur in the time before the fuel disassembles i.e. during the time the fuel is confined on account of its finite mass. A good approximation for the inertial confinement time τ_E is then the time it takes for an ion to move over the distance R, at its thermal speed V_{thi}, taken as the sound speed $\sqrt{kT/m}$. The ICF burn criterion is known as the ρR-criterion, also called the high-gain condition, and is essentially obtained by requiring that almost all the fuel contained in the pellet is indeed burned, i.e. that the number of reactions that take place during the time interval τ_E equals the number of fuel deuterons or tritons. The standard form reads[3]:

$$\rho R \geq 4\sqrt{mkT} < \sigma v >^{-1} \tag{1}$$

where m is the mean ionic mass, the mass density $\rho = nm$, and $< \sigma v >$ is the fusion reaction rate constant. For D-T $\rho R \geq 3g/cm^2$ at $T = 50keV$. The ρR-criterion can also be rewritten in terms of density and confinement time, as $n\tau_E =< \sigma v >^{-1}$. The triple product that results from this puts the reactor requirement typically 10 times higher than what is asked for MCF, a consequence of the inherent inefficiency in assembling the fuel. Please note also that in ICF the term ignition does not have the same meaning as in MCF, as it refers to the condition of efficient α-particle capture, a ρR value of at least $0.3g/cm^2$ being required to slow the α-particles down in the pellet[4].

Since DT-ice has a mass density $\rho = 0.2 g/cm^3$, satisfying the ρR-criterion asks for massive targets, requiring for their heating unattainable amounts of driver energies. An escape from this apparent impasse is however possible. By compression of the pellet, ρ can be increased significantly. An increase by, for instance, a factor of 1000 would lower the energy demand by 10^6, thus bringing it in the range of what is technically achievable. In addition, it is not obvious that the total amount of heat that is needed to bring the fuel to fusion temperatures must be provided by the lasers or beams. It might be enough to ignite a fraction of the pellet and let the fusion energy, thus liberated, heat the rest. The latter requirement is also dictated by considerations of the energy economy of the scheme. It is easy to show that the intrinsic gain G_i of a uniformly heated D-T pellet, defined as the ratio of the energy liberated by fusion to the energy needed to reach the fusion conditions, is at most about 200. The efficiencies in the external systems of the power plant and the low efficiencies of the driver generation, ask for reactor target gains of about 100. Noting that $G = \eta_T G_i$ requires in turn intrinsic gains of about 10^4 to account for a realistic coupling efficiency T of the driver. For inertial confinement to be attractive, it is therefore mandatory to demonstrate that it is possible to burn the whole pellet after bringing just a small fraction to ignition temperature at the densities imposed by the ρR-criterion. The reader is referred to Refs. [2]-[4] for more details on pellet compression and hot spot creation.

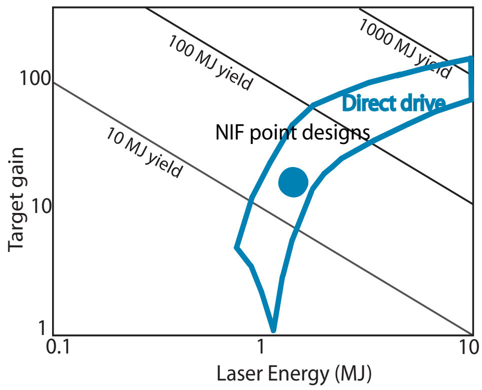

FIG. 1: Expected path of ICF towards achieving ignition and high gain.

Experiments show that satisfying Eq.(1) might be sufficient to achieve the high values of G needed. Figure 1 shows the calculated target gain as a function of direct drive energy [4, 5]. Based on the experimental progress and on the steady advances in system efficiency, it is predicted that ignition should be possible with a driver energy of $0.5 - 1\ MJ$, whereas high gain reactor operation becomes feasible with a $5 - 10\ MJ$ of driver energy. The projected operation point of the US National Ignition Facility (NIF), presently under construction[6] and in which ignition is predicted, is also shown.

III. ICF CONFIGURATIONS

At the heart of an inertial fusion explosion is a target that has to be compressed and heated to fusion conditions by the absorption of energy carried by a driver. For the so-called direct drive, the target consists of a spherical capsule that contains the DT fuel (Fig. 2b). For indirect drive, the capsule is contained within a cylindrical or spherical metal container or hohlraum which converts the incident driver energy into X-rays that then drive the capsule implosion (Fig. 2a). The drivers can be lasers, heavy ion beams or so-called Z-accelerators. The latter consists of a huge array of separate pulsed power devices timed to fire, all to within ten billionths of a second, a current of tens of millions of amperes into two spool-of-thread-sized arrays of 100 to 400 wires, symmetrically positioned with respect to the hohlraum (only one such array is shown in Fig. 2c). The currents vaporize the wires, thus creating a plasma, and produce powerful magnetic fields that pinches this plasma to densities and temperatures sufficient to generate an intense source of Xrays. The main challenge for ICF reactor implementation will be the target manufacturing cost, the repetition rate and target standoff distance at which drivers and windows can be operated and the fusion target chamber construction.

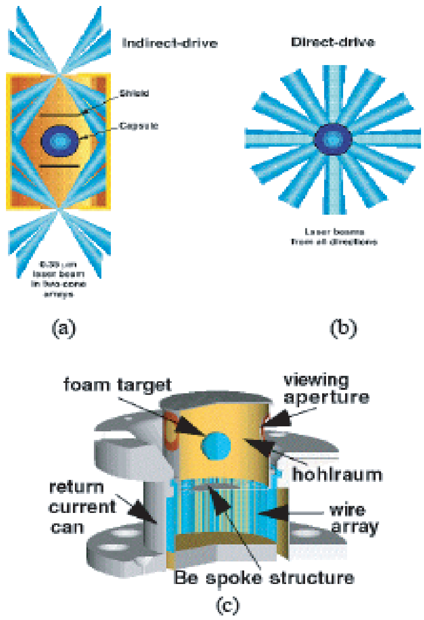

FIG. 2: Geometrical arrangements to implode ICF capsules.

IV. MAGNETIC CONFINEMENT [7, 9]

The Lorentz force makes charged particles move in helical orbits (Larmor orbits) about magnetic field lines. In a uniform magnetic field and in the absence of collisions or turbulence, the particles (better: their guiding centers) remain tied to the field lines but are free to move along them. The distance between the actual particle orbit and the magnetic field line is the Larmor radius r_L. A magnetic field is thus capable of restricting the particle motion perpendicular to the magnetic field but does not prevent particles from moving along the magnetic field. This effect serves as the basis for all magnetic confinement schemes, while at the same time it points to the absolute necessity to cope with the particle losses along the magnetic field (end losses).

The perturbative effect of collisions and turbulence on the transport of particles and energy across the magnetic field can be understood in terms of a simple statistical diffusion process applied here to a cylindrical plasma. Let us first consider Coulomb collisions. The particles suffer collisions with a characteristic collision time τ_c. A collision allows the particle to step across B with a step length equal to r_L. This gives a diffusion coefficient $D \approx r_L^2/\tau_c$. The effect of (electrostatic) turbulence on the other hand can be estimated in a similar fashion. A simple model pictures the particles to be dragged along by the turbulent waves. The step length is now of the order of the wavelength perpendicular to the magnetic field $k_\perp^{-1}$ and the effective collision time is that of the correlation time of the turbulence τ_{corr} , yielding[10] $D \approx 1/(k_\perp^2 \tau_{corr})$. In both cases however, the confinement time is linked to D by means of the simple diffusion relation

$$\tau \approx \frac{a^2}{D} \tag{2}$$

where a is the radius of the plasma, such that in any case high τ requires a large plasma cross-section.

In its motion around a magnetic field line, a gyrating particle constitutes a small current loop of magnetic moment μ that generates a magnetic field that opposes the imposed magnetic field by an amount that is proportional to the kinetic energy contained in the perpendicular particle motion: plasmas in magnetic fields are therefore naturally diamagnetic. The larger the sum of the kinetic energies of all the plasma particles, the lower will be the field. This obviously means that there is a limit to the total energy content ($3nkT$) that a given magnetic field can confine.

The same conclusion is reached by an alternative approach, in which the action of the magnetic field on the confined plasma can be viewed as a balance between the magnetic pressure $B^2/(2\mu_0)$ (μ_0 is the vacuum permeability) and the plasma pressure p according to the relation:

$$p + B^2/(2\mu_0) = constant. \tag{3}$$

The maximum pressure that possibly can be confined at a given B, is thus $B^2/(2\mu_0)$. Stability constraints prevent however the attainment of this maximum and the pressure thus reaches at most a fraction β (beta) of its theoretical limit[11, 12]. A large value of B is therefore the key to achieve large values of $p = nkT$.

From what we just have seen, it is to be expected that the fusion triple product in devices without end losses will increase with plasma cross section (Eq. 2) and magnetic field pressure (Eq. 3). Such a dependence is substan-

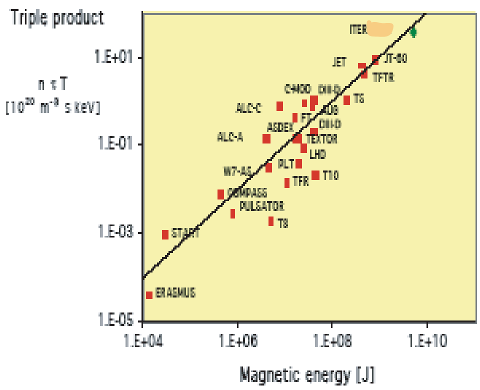

FIG. 3: $n\tau_E T$ values reached by MCF devices versus the magnetic energy stored in their plasma volume.

tiated in Fig. 3, showing the $n\tau_E T$ values experimentally achieved over 30 years of research in a large number of toroidal magnetic fusion devices as a function of $E_{mag} = B^2/\mu_0 V$, the total magnetic energy stored in the plasma. The scatter in the data is caused by differences in configuration as well as in secondary engineering parameters. This graph predicts that magnetic fusion will achieve reactor grade $n\tau_E T$ values in the projected ITER device (diamond).

V. MAGNETIC CONFIGURATIONS WITH END LOSSES

One could in principle conceive a magnetic confinement machine that consists of a long solenoid of length L in which particles are confined radially but flow out axially. By analogy with the ICF-case, one could define an effective energy confinement time:

$$\tau_E = \frac{L}{V_{thi}}. \tag{4}$$

For $L = 1\ km$, τ_E equals about $10^{-3}\ s$ at $T = 15\ keV$, meaning that burn is possible for $n > 2 \times 10^{23} m^{-3}$. The pressure corresponding to these n and T values requires a confining field $B = 50\ T$. It is therefore clear that the end losses have to be curtailed in a fusion reactor. One way to achieve this is through an increase of the magnetic field strength at each end of the solenoid. The gyrating particles will then be repelled from these areas with higher field strength, which thus effectively act as "magnetic mirrors". The reflection is due to the socalled grad-B force:

$$F_z = -\mu \frac{\partial B_z}{\partial z}, \tag{5}$$

where $\mu = \frac{1}{2}mv_\perp^2/B$ is the magnetic moment of the particle. It can be shown that is an adiabatic invariant, meaning that its value does not change along the motion. The motion of a particle in such a mirror can then also be described by means of the two conservation relations

$$mv_\perp^2 + mv_z^2 = C^{st}, \tag{6}$$

and

$$\mu = \frac{\frac{1}{2}mv_\perp^2}{B} = C^{st}. \tag{7}$$

During the motion towards a higher field, $v \perp$ increases and v_z decreases. At sufficiently high values of B, v_z can even be reduced to zero, i.e. the particle reflects. Although the end losses can be significantly reduced in a mirror[13], the confinement of such a device proved to be too low and mirror machines have almost completely disappeared from the fusion scene.

VI. TOROIDAL PARTICLE CONFINEMENT.

An obvious recipe for the elimination of the end losses is to close the magnetic field lines on themselves. This

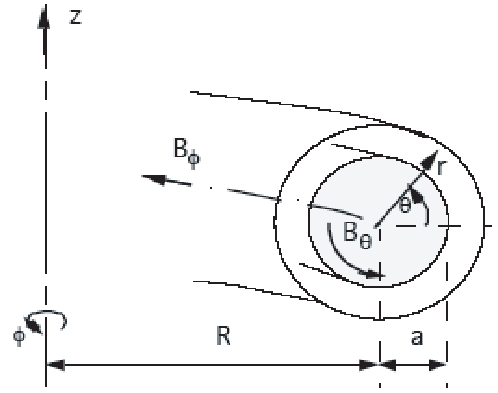

FIG. 4: Coordinates and fields in a toroidal system.

can for instance be done by aligning the field producing coils along a circumference of radius R, thus creating a toroidal magnetic field, B (see Fig. 4) having a gradient in the direction of R. During their motion along the toroidal field lines the plasma particles experience a radially outward directed force F_R which is the sum of a centrifugal force $mv_{//}^2/R\vec{e_R}$ and a grad-B force $\frac{1}{2}mv_\perp^2/B\vec{e_R}$. As a result, a drift motion, the so-called toroidal drift v_D, occurs that is traverse to both the field and the field gradient and is given by:

$$\vec{v_D} = m\frac{v_{//}^2 + \frac{1}{2}v_\perp^2}{q_c R B^2}\vec{e_R} \times \vec{B}. \tag{8}$$

Averaging over a Maxwellian, the value of the toroidal drift becomes

$$v_D = \frac{r_L}{R}V_{th}. \tag{9}$$

Because of the dependence on charge q_c, electrons and ions experience drifts in opposite directions, giving rise to the creation of an electric field. The latter than causes both electrons and ions to drift together radially outwards, thus shattering our hopes of creating the ideal confinement system.

The catastrophic effect of the toroidal drift can be avoided by twisting the magnetic field lines helicoidally[8, 14]. One uses the term rotational transform to characterise the twisting, which gives rise to a poloidal field component B_θ . The amount of rotational transform is measured by the ratio B_θ/B_ϕ, or by the rotational transform angle $\iota = 2\pi/q$ where q, the safety factor, is defined as:

$$q = \frac{rB_\phi}{RB_\theta}. \tag{10}$$

If one follows a given field line many times around the torus a closed flux tube is mapped, a so-called magnetic surface. Surfaces pertaining to different field lines form a set of nested surfaces around the torus axis. It should be noted that the rotational transform angle is in general different from surface to surface: the configuration therefore possesses magnetic shear, a property which is quite effective against large scale plasma instabilities.

By considering the trajectory of a single particle (with high enough velocity $v_{//}$ along the magnetic field), it is easy to show that the helical twist can compensate the toroidal drift. It suffices to show that, even in the presence of v_D, the trajectory of a charged particle is a closed orbit. Without the toroidal drift, the trajectory of the guiding centre of a particle coincides with a field line, such that its projection on a meridian plan (coordinates x and y) is a circle which the particle describes with an angular frequency $\omega = v_{//}B_\theta/(aB_\phi)$. Including the toroidal drift, the projected trajectory is found from:

$$\frac{dx}{dt} = \omega y + v_D \tag{11}$$

$$\frac{dy}{dt} = -\omega x.$$

the solution of which is a circle which is displaced with respect to the projection of the field line such that the maximum distance between the orbit and the magnetic surface equals:

$$d = 2\frac{v_D}{\omega} \approx qr_L. \tag{12}$$

We therefore conclude that a toroidal system with rotational transform can indeed confine particles. The price to be paid to get rid of the end losses is an excursion of the particles away from a magnetic field line that is larger by the factor q than the Larmor radius. As this excursion turns out to be the step length for collisional transport, one sees that it is of great advantage to work with as low a q as possible, i.e. with the highest possible helical twist. The maximally allowed amount of twist will result from stability considerations[11, 12]. Note also that the rotational transform provides a conductive path between the top and bottom zones of opposite charge polarity: the currents that thus flow are called the Pfirsch-Schlter currents. As these currents meet some resistance, the vertical electric field can not completely be short-circuited.

The dynamical behaviour of a plasma in a magnetic field is more intricate than just being the sum of the motions of the individual particles. One can show that a perfectly conducting plasma is capable of dragging the magnetic field lines along during its macroscopic motion. One talks about field lines that are frozen into the plasma. In this sense, we can conclude that the particle motion described earlier pertains to motion with respect to a fixed magnetic field, i.e. where any macroscopic motion of the field lines (and of plasma) is prevented. In a tokamak, we will see that the radial force F_R ($= 4\pi^2 a^2 p$ when summed over all Maxwellian plasma particles), has to be compensated by means of an additionally applied vertical magnetic field. This motion of plasma and field plays also an important role in the so-called pressure driven instabilities of the interchange and ballooning type.

VII. THE TOKAMAK[10, 16]

A tokamak is a toroidal device in which the poloidal magnetic field is created by a toroidal current I_p flowing through the plasma. Figure 5 gives a schematic diagram of a tokamak. A strong toroidal magnetic field is generated by a toroidal field coil system. The toroidal current is induced by means of a transformer. The plasma itself forms the secondary winding of the transformer, the primary being wound on an iron core.

The toroidal geometry of the plasma leads to two hoop forces which are both in the direction to expand the plasma ring. The first of these forces results from the natural tendency of a current loop to expand in an effort to lower its magnetic energy. The second force is the resultant F_R of the sum of centrifugal and grad-B forces experienced by the individual particles during their motion along the field lines.

Both these forces can be compensated by providing a vertical magnetic field B_v, that interacts with the toroidal current to give an inward force. The required magnitude of this field is:

$$B_v = \frac{\mu_0 I_p}{4\pi R_0}[(\ln\frac{8R_0}{a} + \frac{l_i}{2} - 1.5) + \beta_p]. \tag{13}$$

In this expression, β_p is the ratio of the mean plasma pressure to the poloidal magnetic field pressure and describes that part of vertical field that is needed to compensate F_R. l_i is the internal inductance per unit length of the current loop and, together with the external inductance given by the other terms between the round brackets, sets the amount of field needed to balance the current force contribution.

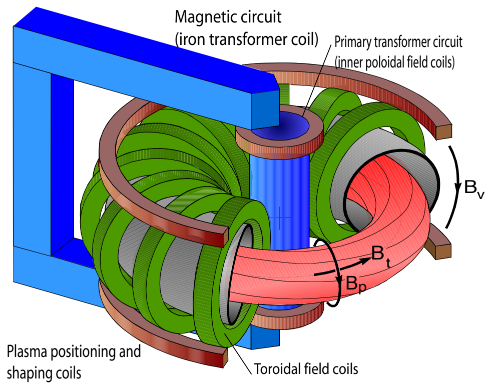

FIG. 5: Schematic diagram of a tokamak.

If the applied vertical field is spatial non-uniform and increases with major radius, the plasma is found to be vertically unstable. Such a vertical field shape is e.g. mandatory when, in an attempt to increase the plasma pressure, the plasma is pushed as much as possible to the high field side, thus creating a D-shaped plasma, i.e. having elongation and triangularity. An externally applied horizontal magnetic field Bh can then be used to maintain the plasma well centred. Both the horizontal and vertical position control is in all modern tokamaks achieved by means of feedback controlled vertical and horizontal magnetic field systems. The combination of the above fields can generate an equilibrium tokamak configuration. Whether this equilibrium will be stable or unstable can be found from a stability analysis. A tokamak plasma has essentially two origins of instability, i.e. two energy sources for the excitation of oscillations: the magnetic energy of the plasma current and the plasma thermal energy. At high pressures, these sources start to interact with each other, but at low pressure they can be studied separately. The poloidal field magnetic energy excites helical instabilities, named kink instability and tearing instability, while the thermal energy excites flute (or interchange) modes and ballooning modes.

VIII. THE MCF REACTOR

The tokamak is the most studied and most advanced fusion machine to data and is the most likely system to be converted into a reactor. Even when the confinement time of toroidal configurations still lacks a quantitative first-principle derivation on account of the intricate nature of plasma turbulence, important progress has been achieved through an empirical approach[17], akin to windtunneling, and has allowed to find the most important engineering parameters affecting confinement and has brought the attainment of the burn condition in a tokamak at hand.

It is however not clear today whether the tokamak is the optimal reactor concept. Some alternative approaches are therefore being pursued, in the first place the Stellarator[18], and, on a more exploratory level, such devices like the Reversed Field Pinch[19], the Spheromak[20] and the Field Reversed Configuration[21]. It is however appropriate here to illustrate on the example of the tokamak which are the main developments required on the way to the reactor.

A first series of issues has to do with economical reactor operation and the likelihood to achieve unit sizes that are acceptable in power output, in physical volume or in cost of electricity. Mechanical endurance and duty cycle considerations require the burn to be sustainable for a long, in principle unlimited time. There are two problems here. Firstly, as long as its plasma current is generated by induction, the tokamak is a pulsed system. One might therefore have to develop alternative ways to generate I_p, known as current-drive methods, or possibly switch to an alternative confinement scheme like the stellarator.

Secondly, it is clear that sudden termination of the discharge, known as disruption, should be avoided (another plus for the stellarator) or by a burn quench due to ash or impurity accumulation.

This last problem falls under the heading heat and particle removal and is a prime object of present day's research. A reactor will have to exhaust power and particles associated with the thermalisation of the 3.5 MeV alpha particles. The power leaves the plasma in the form of radiation or of kinetic energy of the escaping particles. The problems and solutions will differ depending on how the plasma is limited. Direct contact of the plasma and the material wall is avoided because unavoidable imperfections in the magnetic configuration and motions of the plasma column might lead to concentrated heat deposition on areas that are difficult to control and cool.

To this end, a specially monitored, suitably clad and cooled piece of wall, somewhat protruding from the main wall, is often used to intercept the escaping particles. This element is called limiter. The limiter's exposed surface should be large enough to avoid too large power fluxes and it is therefore indicated to use a toroidal (or belt) limiter that runs around the circumference of the torus. The magnetic surface that touches the inner most part of the limiter is called the last closed flux surface (LCFS). It is also possible to exhaust the escaping particles into a separate chamber before they actually reach a material wall. By means of extra magnetic coils, the magnetic configuration inside the containing vessel can be divided in two zones, separated by a so-called separatrix (= LCFS). Inside the separatrix there exist the desired nested and closed magnetic surfaces. A particle escaping from this inner zone towards the outside (into the so-called scrapeoff layer) meets field lines that convey it to a target plate in the exhaust chamber, which can be situated quite far from the plasma boundary at the separatrix. When the extra field coils consist of conductors that are concentric with the plasma current, the configuration is called an axisymmetric or poloidal-field divertor. The point where the poloidal field is zero is called the X-point. The limiter or the divertor target plates are heated by the incoming exhausted power and bombarded by the escaping particles. As a result, material is released from their surfaces which can reach the plasma in the form of neutral particles, capable of deep penetration before being ionised. As such particles are impurities that can cause a lot of radiation loss from the plasma and in addition lead to fuel dilution, it is very important to (i) reduce the power density to the targets to levels that can be handled by state-of-the-art cooling techniques and (ii) decrease the kinetic energy of the incoming particles below the threshold energy at which target damage occurs. A special category of escaping particles are the helium atoms produced in the fusion reactions: care should be taken that these leave the plasma promptly and are not given the chance to reenter the discharge as impurities.

Providing the needed vacuum enclosure, the first wall is probably the most critical reactor component, as it is the target of very intense radiation from the plasma (14 MeV neutrons, energetic neutral particles produced by charge exchange, photons of various energies). Its mechanical strength will be weakened by lattice damage and swelling, by wall erosion through sputtering and by temperature excursions. In addition, neutron induced transmutation reactions can render the wall radioactive. Based on these extreme operational conditions, it is presently estimated that the time integrated neutron flux through the first wall will have to be lower than about 18 $MW - y$ per m^2. Upon reaching this limit, the first wall will have to be replaced[22]. Solving the heat and particle removal issue and finding adequate first wall materials are the prime tasks of present day's fusion research, presenting an equally large challenge to the tokamak and its possible alternativesMany of these problems point to the urget need for an irradiation facility for fusion materials, such as IFMIF[23].

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