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Fusion energy

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Summary. — This paper presents an overview of the main challenges that fusion research is facing on the road to a demonstration power plant. The focus is on magnetic confinement fusion. Most of the challenges are being addressed in the context of the ITER construction and exploitation. These include the demonstration of high fusion gain regimes of operation, the management of high heat and particle loads and the integration of the main technologies of a fusion power plant. In preparation of DEMO, reliable solutions for the breeding blanket and neutron resistant materials have to be developed.

1. – Introduction. Fusion in the context of energy technologies

In the quest for an energy system that meets simultaneously the needs of sustainability, security of supply and economic competitiveness Europe is pursuing several technologies since a robust energy strategy has to rely on a portfolio of different energy sources that include nuclear, solar, wind and hydropower [1].

In this context, fusion can play an important role since it has a number of advantages:

– It is virtually unlimited. For the deuterium-tritium reaction the fuel in sea-water could supply the Earth (at the present consumption rate) for several million years.

– It does not produce greenhouse gases.

– It is intrinsically safe.

TABLE I. – *Fusion reactions.*

$D + T$	$\rightarrow {}^4He\ (3.5\,MeV) + n\ (14.1\,MeV)$	
$D + D$	$\rightarrow T\ (1.01\,MeV) + p\ (3.02\,MeV)$	50%
	${}^3He\ (0.82\,MeV) + n\ (2.45\,MeV)$	50%
$D + {}^3He$	$\rightarrow {}^4He\ (3.6\,MeV) + p\ (14.7\,MeV)$	
$T + T$	$\rightarrow {}^4He + 2n + 11.3\,MeV$	
${}^3He + T$	$\rightarrow {}^4He + p + n + 12.1\,MeV$	51%
	$\rightarrow {}^4He\ (4.8\,MeV) + D\ (9.5\,MeV)$	43%
	$\rightarrow {}^5He\ (2.4\,MeV) + p\ (11.9\,MeV)$	6%(1)
$p + {}^6Li$	$\rightarrow {}^4He\ (1.7\,MeV) + {}^3He\ (2.3\,MeV)$	
$p + {}^7Li$	$\rightarrow 2{}^4He + 17.3\,MeV$	20%
	$\rightarrow {}^7Be + n - 1.6\,MeV$	80%
$D + {}^6Li$	$\rightarrow 2{}^4He + 22.4\,MeV$	
$p + {}^{11}B$	$\rightarrow 3{}^4He + 8.7\,MeV$	

– It is environmentally responsible. The primary reaction does not produce radioactive materials. It does produce neutrons that activate the reaction chamber. However, with a proper choice of materials, radioactivity decays in a few tens of years and after a century they can be recycled in a new reactor.

Several fusion reactions can in principle be exploited (see table I). The cross section σ and the reactivity $\langle \sigma v \rangle$ (the product of the cross section and the particles relative velocity averaged over the distribution functions of the reacting species) are shown in fig. 1 for the DT, D^{3}He and DD reactions.

The reaction with the largest cross section is the reaction between two hydrogen isotopes: deuterium and tritium. The DT reaction produces a 3.5 MeV alpha particle and a 14 MeV neutron. To make fusion energy with the DT reaction we need to burn deuterium and lithium. There is plenty of deuterium in sea-water (about 35 mg per litre [2, 3] —making fusion with deuterium alone would supply the Earth, at the present consumption rate, for several billions years!). On the contrary, tritium does not exist on Earth. It is mildly radioactive (it undergoes beta decay releasing an electron with a maximum energy of 18.6 keV) with a half-life of 12.3 years. Therefore, it has to be

(0) The ^{5}He isotope decays in 7×10^{-22} s in a neutron plus an alpha particle.

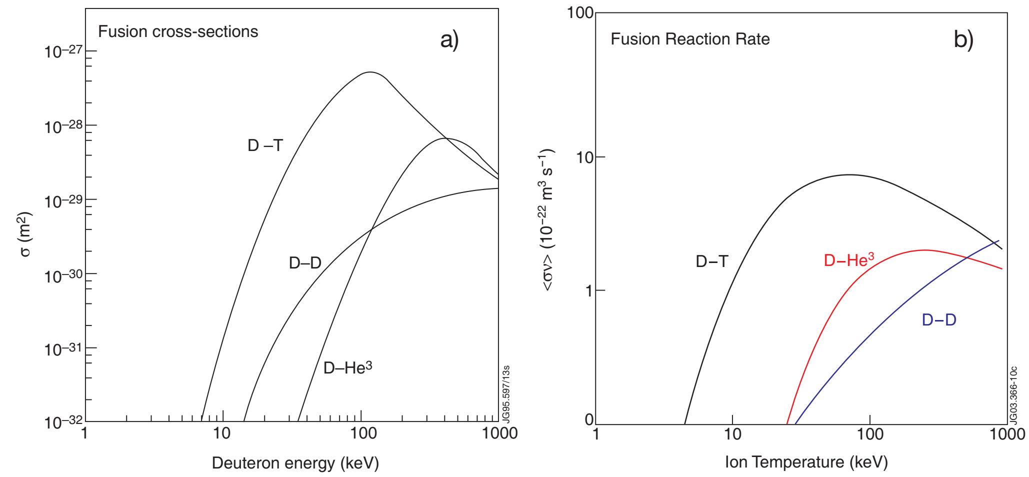

Fig. 1. – Fusion cross sections and fusion reactivity for the three main fusion reactions. The cross section is plotted as a function of the energy of the deuteron impinging on the target at rest.

produced inside the reactor and this is where lithium comes into play. The reaction between a neutron and a lithium nucleus produces a He nucleus together with a T nucleus that can be extracted and re-circulated in the reactor.

In order for D and T to come at sufficiently close distance for fusion to occur they need to overcome their mutual Coulomb repulsion. This can be achieved by heating the DT gas to sufficiently high temperature ($\sim 20\,\text{keV}$ or 200 millions of $^\circ$C) in such a way that the ions in the tail of the Maxwellian distribution function provide a significant number of reactions. Matter at these temperatures is in the *plasma* state. The electrons are no longer bound to the nuclei and the system becomes the superposition of two gases of negatively charged electrons and of positively charged ions. To confine plasmas at these temperatures two main methods have been successfully employed (see fig. 2).

- The first method is magnetic confinement that will be mostly covered in this lecture. In this case we can take advantage of the charged nature of the plasma constituents to confine them using intense magnetic fields ($\sim$ 100000 times larger than the average Earth magnetic field). In the presence of a magnetic field particles move along field lines as a train on a rail and are therefore confined in the plane perpendicular to the magnetic field. In order to confine plasmas also in the third direction the magnetic field lines are wound in such a way as to form a set of nested toroidal surfaces called magnetic surfaces, with each line lying on a magnetic surface. The combination of intense magnetic fields ($\sim 5\,\text{T}$) and toroidal geometry enables plasma confinement.

- The second method is inertial confinement (see ref. [4]). In this case a sphere of solid fuel is irradiated with electromagnetic waves or energetic particles. The surface

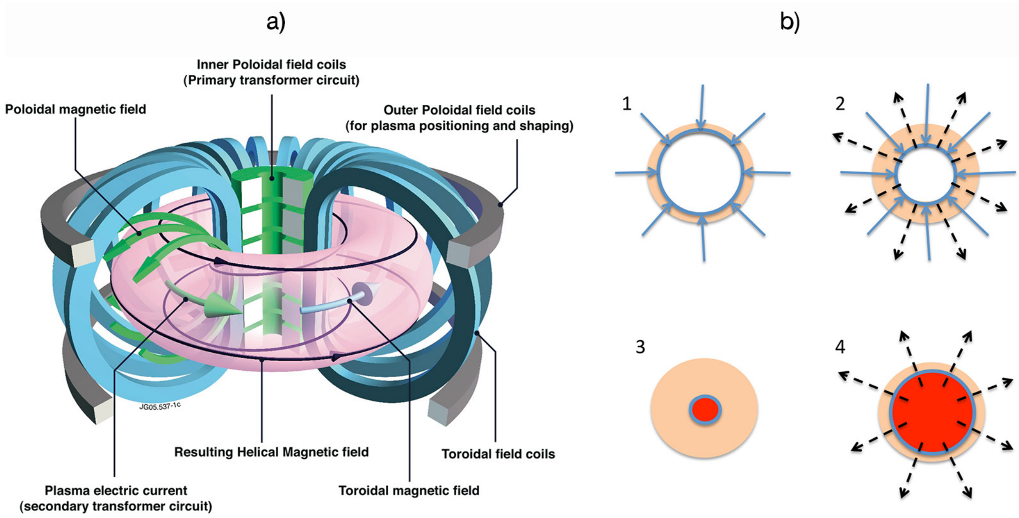

Fig. 2. – (a) Magnetic confinement geometry illustrated for the tokamak configuration —see text (courtesy EUROfusion). (b) 1: Inertial confinement geometry. The shell target (blue circle) is irradiated with the driver beams (blue arrows) and a spherical cloud of ablated plasma is formed. 2: The shell implodes as a consequence of ablation. 3: A hot spot of compressed fuel is formed where fusion reactions take place. 4: The fuel is burned and an explosion occurs.

layer is ablated and its particles are ejected in the radial direction. This produces a radial compression that, if sufficiently large, can burn a large fraction of the fuel.

It is important to stress that with both methods plasmas of thermonuclear interest can be already routinely produced. In addition, a significant amount of controlled fusion reactions have been already obtained. In a dedicated campaign on the JET machine in 1997 [5] up to 16 MW of fusion power has been obtained in transient conditions to be compared with about 25 MW of power injected in the reaction chamber. Nevertheless, making a fusion power plant requires the solution to a number of other challenges that will be discussed in the following.

In 2012 the European Commission requested a Roadmap to fusion Energy in order to understand if fusion can start playing a role in the electricity production on the time scale of 2050 used for the Energy Roadmap of the European Union. The Fusion Roadmap [6] (an update has been recently produced by EUROfusion [7]) was based on the following assumptions:

– A program focussed around the priorities and organized in eight missions.

– A pragmatic approach —build on the shortest possible time scale a device capable of breeding tritium and produce electricity.

– Early involvement of industry.

– Full exploitation of the opportunities arising from international collaborations.

The Roadmap has been the reference document for the EURATOM activities in fusion for Horizon 2020.

The Roadmap foresees two main facilities, ITER and DEMO:

– ITER will demonstrate burning plasma operation, *i.e.* operation with dominant plasma heating produced by the fusion generated alpha particles. ITER is also developing a large part of the technologies needed in a fusion power plant. However, ITER will neither produce electricity nor will breed the tritium it uses.

– DEMO will be the first demonstration fusion power plant. In addition to what ITER will accomplish, it will produce a net electricity output and will be self-sufficient for the production of tritium.

Both ITER and DEMO are machines based on a specific magnetic confinement configuration called *tokamak* [8] and schematically shown in fig. 2(a). The plasma is characterized by a doughnut shape (*torus*) with major radius R_o and minor radius a. The magnetic field is the superposition of a toroidal component B_f (mostly generated by a set of external coils) and a poloidal component B_p (mostly produced by a current flowing in the plasma itself in the toroidal direction). The combination of a toroidal and a poloidal magnetic field produces magnetic field lines that lay on toroidal surfaces called magnetic surfaces. The winding pitch of the magnetic field line on each magnetic surface is described by the *rotational transform* (the number of turns made in the poloidal direction for a single turn in the toroidal direction). The inverse of the rotational transform is the *safety factor*. The existence of a rotational transform is essential for plasma confinement. Without a rotational transform ions and electrons, under the effect of magnetic field curvature and spatial in-homogeneity, would drift vertically in opposite directions producing an electric field that, combined with the toroidal magnetic field, would push the plasma outwards in the radial direction in a very short time.

After a confined plasma is established, it is heated up to temperatures of 10–20 keV by auxiliary heating systems until the fusion generated alpha particles become the dominant heating source. The heat generated in this way is transported from the center of the plasma to the edge (see sect. **2**) and removed from the reaction chamber. Similarly, the He ashes produced after the fusion alphas have transferred all their energy to the plasma need to be continuously removed to avoid poisoning the fuel (dilution). Plasma exhaust takes place at a special location, called *divertor*, in the reaction chamber (usually a niche at the bottom) sufficiently remote from the hot plasma such that the heat and particle removal can take place without disturbing the dynamics of the plasma inside the *magnetic separatrix*. The separatrix defines a sharp boundary between the hot plasma region (inside) and the plasma edge region in contact with the chamber wall and the divertor and therefore at much lower temperature.

In the following we review the main challenges of fusion research. The focus will be on magnetic confinement although a number of challenges are common also to inertial confinement.

2. – Demonstration of regimes with a high energy gain

The first challenge is the demonstration of regimes in which the amount of fusion power absorbed in the plasma is much larger than the external heating power (burning plasma conditions). The alpha particles produced in fusion reactions are charged and therefore can be confined by the same magnetic field that confines the plasma. They release their energy to the plasma via collisions. In burning plasma conditions the alpha particle heating can almost entirely maintain the high plasma temperatures. In ITER the target is a plasma in which alpha heating (100 MW) is twice the external heating (50 MW). Since the 14 MeV neutrons carry four times more energy than the 3.5 MeV alpha particles the fusion power produced in the form of neutrons in ITER is 400 MW. Thus, the total fusion power is 100 MW + 400 MW = 500 MW and the fusion gain Q (given by the ratio between the fusion power and the external power injected in the reaction chamber) in ITER is Q = 500 MW/50 MW = 10. For comparison, the JET achievement quoted above (16 MW of fusion power with 25 MW of externally injected power [5]) corresponds to $Q \sim 0.67$.

Why high gain has not been achieved so far? The problem is associated with the conduction losses due to the small-scale turbulence destabilized by the free-energy sources always present in a confined plasma (the gradients of temperature and density). Radiation losses are also present although usually much smaller. Losses tend to cool down the plasma and must be balanced by the heating power. What we know from theory and experiments is that, at fixed density, temperature and magnetic field, the power lost via conduction is at most linearly increasing with the machine radius R [9] whereas the fusion power increases as R^3. Therefore the solution that has been pursued has been to make the machine size larger.

The argument above can be put on a quantitative basis as follows. For the sake of simplicity we will assume that density and temperature are constant in space. The power generated by fusion reactions is given by

$$P_{fus} = 17.6\,\mathrm{MeV}\, n_D n_T \langle \sigma v \rangle V, \tag{1}$$

where n_D is the particle density of the deuterium nuclei (number of deuterium nuclei per unit volume), n_T is the particle density of the tritium nuclei, $\langle \sigma v \rangle$ is the Maxwellian fusion reactivity and V is the plasma volume. The Maxwellian reactivity is a function of the temperature of the reactants [10].

The power P_{cond} lost by the plasma through conduction is due to small-scale turbulence and can be quantified in terms of the energy confinement time τ_E

$$P_{cond} = W/\tau_E, \tag{2}$$

where $W = (3/2)\ (n_e T_e + \Sigma_j n_j T_j)\ V$ is the internal energy of the plasma with the sum extended to all the ion species and $T_e(T_i)$ the electron (ion) temperature. The energy confinement time is the characteristic time for the plasma to cool down once the heating sources are switched off. It has nothing to do with the time plasma is confined. In ITER τ_E is a few seconds whereas the plasma can be confined for hundreds of seconds.

The power lost by the plasma through radiation is mostly associated with Bremsstrahlung and is given by

$$P_{brem}\,(\mathrm{MW}) = 1.69 \times 10^{-4} Z_{eff} n_e\,(10^{20}\,\mathrm{m}^{-3})^2 T_e\,(\mathrm{eV})^{1/2} V\,(\mathrm{m}^3) \tag{3}$$

with n_e the particle density of the electrons and $Z_{eff} \equiv \Sigma_j\, Z_j^2\,(n_j/n_e)$ the average charge of the plasma ions.

Since the plasma must be locally neutral the electron density n_e must be equal to the sum of the ion densities n_j times their charge Z_j

$$n_e = \Sigma_j Z_j n_j. \tag{4}$$

The power balance equation in stationary conditions dictates that the fusion power released in the form of alpha particles ($3.5\,\mathrm{MeV}/17.6\,\mathrm{MeV} \sim 20\%$ of the total) plus the power P_{aux} injected from external sources must be equal to the power lost by conduction and radiation

$$(3.5/17.6)P_{fus} + P_{aux} = P_{cond} + P_{brem}. \tag{5}$$

The fusion gain $Q = P_{fus}/P_{aux}$ becomes infinite for $P_{aux} = 0$, meaning that the fusion reactions are self-sustained. This is the so-called ignition condition.

For a pure deuterium and tritium plasma ($n_D = n_T = n_e/2$) with equal electron and ion temperatures, the ignition condition can be written as

$$n_e\,(10^{20}\,\mathrm{m}^{-3})\tau_E\,(\mathrm{s}) = 3.42 \times 10^{-3}\,T\,(\mathrm{keV})\Big/\left[\langle\sigma v\rangle(10^{-20}\,\mathrm{m}^3/\mathrm{s}) - KT\,(\mathrm{keV})^{1/2}\right] \tag{6}$$

with $K = 3.82 \times 10^{-4}\,\mathrm{m}^3\,\mathrm{s}^{-1}\,\mathrm{keV}^{-1/2}$. This is the Lawson criterion (fig. 3(a)). It expresses the requirement in terms of the product of density and confinement time to achieve ignition as a function of the plasma temperature.

The right-hand side of the Lawson criterion is a function of the temperature only with a minimum corresponding to $n_e\tau_E \approx 1.6 \times 10^{20}\,\mathrm{m}^{-3}\mathrm{s}$ at about $25\,\mathrm{keV}$. For temperatures approaching $4.3\,\mathrm{keV}$ the $n_e\tau_E$ product tends to infinity. This is the so-called ideal ignition temperature and represents the minimum temperature below which the alpha particle power is smaller than Bremsstrahlung. Around the optimal temperature Bremsstrahlung can be neglected and we can approximate the fusion reactivity as $\langle\sigma v\rangle \sim 10^{-24} T\,(\mathrm{keV})^2\mathrm{m}^3/\mathrm{s}$. Upon substituting in eq. (6) we can determine a condition on the triple product $n_e\tau_E T \sim 3.42 \times 10^{21}\,\mathrm{m}^{-3}\mathrm{s}\,\mathrm{keV}$.

The fulfillment of the Lawson criterion is the main goal of fusion experiments and is mostly related with the achievement of high confinement times. Fusion plasmas exhibit the transition to self-organized states in which turbulence is quenched by the formation of sheared flows and a “transport barrier” is locally formed. The most common (and easily

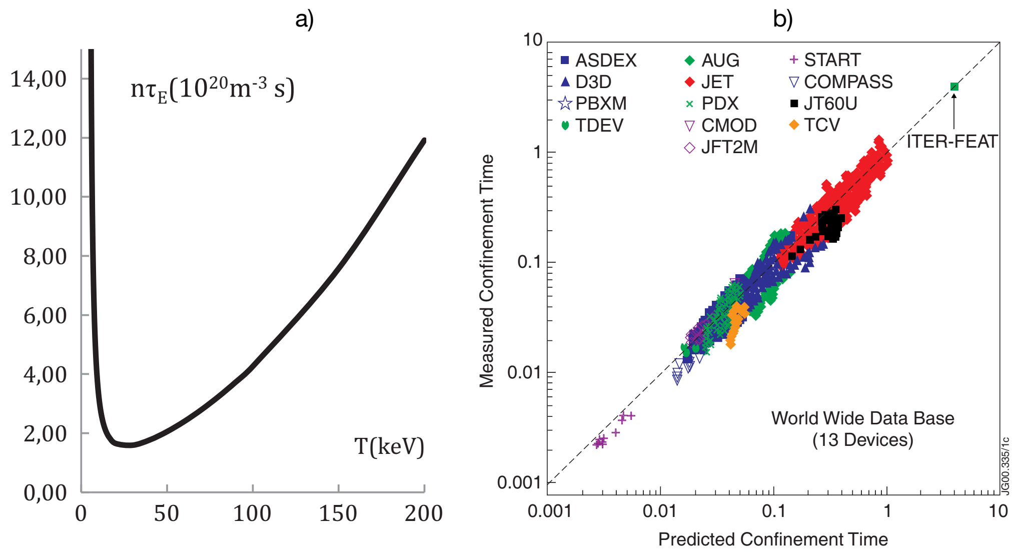

Fig. 3. – (a) Lawson criterion $n\tau_E$ *vs.* temperature for a pure homogeneous DT plasma; and (b) measured energy confinement time *vs.* the prediction of the ITER98y2 scaling (from ref. [9], by courtesy of EUROfusion). ITER-FEAT was the name of ITER at the time of ref. [9].

reproducible) of these self-organized states is the so-called H-mode and is characterized by a transport barrier located at the magnetic separatrix. Unfortunately, the prediction of τ_E from first principle models of plasma turbulence still bears significant uncertainties. Thus τ_E is extrapolated from the experimental values obtained in present day machines that cover a range of three orders of magnitudes (from $\sim$ms values obtained on START to $\sim$s values obtained on JET). A careful analysis of the experimental data has produced the following scaling law for the energy confinement time in H-mode [9]:

$$(7)\quad \tau_{ITER98(y,2)}\ (\mathrm{s}) = \\ 0.0562 I_p\ (\mathrm{MA})^{0.93} B\ (\mathrm{T})^{0.15} P\ (\mathrm{MW})^{-0.69} n\,(10^{19}\ \mathrm{m}^{-3})^{0.41} M^{0.19} R_o\ (\mathrm{m})^{1.97} \epsilon^{0.58} \kappa^{0.78},$$

where I_p is the plasma current, B the toroidal magnetic field, P the heating power, n the line averaged plasma density, M the average ion mass, $\epsilon \equiv a/R$ the inverse plasma aspect ratio and κ the plasma elongation. The extrapolation to ITER gives a value of 4.3 s, about three times larger than the upper limit of τ_E in JET. The estimated value of the triple product on ITER (with $n = 10^{20}\,\mathrm{m}^{-3}$ and $T = 20\,\mathrm{keV}$) is $n_e\tau_E T \sim 8.6 \times 10^{21}\,\mathrm{m}^{-3}\mathrm{s\,keV}$ in line with the requirements to achieve a fusion gain $Q = 10$. It should be noted that although H-mode conditions can be easily reproduced in all the tokamaks with a magnetic separatrix, the H-mode barrier formation requires a minimum power to be transported across the separatrix by conductive losses that is at the moment difficult to estimate both theoretically and experimentally.

In the case of inertial fusion a criterion similar to the Lawson criterion can be derived. In this case the fuel is confined by its own inertia and we can define a confinement time $\tau_{ICF} \sim R_{fuel}/C_s$, where R_{fuel} is the radius of the compressed fuel and C_s the sound velocity. The request that the confinement time is longer than the fusion reaction time $\tau_{fus} = 1/(n\langle\sigma v\rangle)$ yields a condition on the product ρR_{fuel}, (with ρ the fuel mass density) that is analogous to the Lawson parameter $n\tau_E$. For typical ICF parameters the product ρR_{fuel} must be larger than 2–3 g/cm^2 [4].

Part of the challenge of demonstrating regimes with a high energy gain is represented by the need of controlling macroscopic plasma instabilities [11]. Broadly speaking we can distinguish two classes of instabilities:

- Instabilities that lead to a sudden loss of confinement (disruptions). These must be avoided (operating far from the stability threshold), prevented (by deploying appropriate control systems) or, in case they occur, mitigated (*e.g.* by the injection of highly radiating elements) to avoid localized thermal loads on the plasma facing components and the electromagnetic loads on the conductive structural components inside the vacuum chamber.

- Instabilities that produce a minor redistribution of the plasma and can be beneficial in facilitating the ejection of impurities. Instabilities belonging to this class are benign provided their amplitude is kept sufficiently small through appropriate control methods. Examples of instabilities to be controlled are edge localized modes (ELM) or sawtooth oscillations. In both cases the instability takes the form of a periodic relaxation that induces plasma redistribution (at the edge and at the centre, respectively). Several methods have been developed to control these instabilities. Magnetic perturbation coils producing 3D fields or injection of frozen deuterium pellets have been shown to suppress or mitigate ELMs. Millimetre wave injection is a reliable method to control neoclassical tearing modes. These control methods have been demonstrated in present experiments and now await a final confirmation in ITER.

In addition ITER will break new ground in the investigation of collective alpha particle effects, namely the destabilization of modes driven by a population of energetic particles [12]. Our theoretical understanding shows that these modes should lead at most to minor plasma redistribution for the operational scenarios investigated during the first ITER phase but their role for the operations in fully steady-state scenarios is expected to be substantial.

Finally, a specific challenge is related with the achievement of a fully steady state. This is in principle the target of any energy technology that has to provide electricity to the grid. An axisymmetric equilibrium requires a net plasma current to flow in the toroidal direction in order to produce a magnetic field with a rotational transform. The plasma current is produced via magnetic induction by a variable current flowing in a cylindrical solenoid (CS) inserted into the central hole of the torus. The CS acts as the primary winding of a conventional transformer in which the plasma is the secondary winding.

Since the plasma current has to flow always in the same direction (no alternate current operation are foreseen) the current on the primary has to monotonically change starting from the maximum current the central solenoid can bear. If we assume that this current is in the positive direction, the central solenoid current first crosses the zero value and then flows in the negative direction until the maximum current (in the opposite direction) is again reached. Beyond this value it is no longer possible to induce current in the plasma. For axisymmetric configurations there are two solutions to this problem:

- We abandon the idea of continuous plasma operation and, after the central solenoid has completed the flux swing, we stop the plasma and recharge the transformer. This requires some energy storage to be available. Pulsed operation could produce fusion power for a few hours with a recharging time of about 20 minutes. This solution is simple to implement from the point of view of plasma operation but may induce cyclic fatigue in some component.
- The plasma current is generated through a combination of the current self generated by the plasma (the so called bootstrap current) [8] and external current drive systems [13]. These regimes have been demonstrated but are not fully qualified and require a sophisticated control capability of the radial profiles of pressure and current density. ITER has the goal of demonstrating their viability for a reactor.

There is a third possibility: abandon axisymmetric equilibria. In this case the rotational transform can be produced through a set of external helical coils. This would avoid entirely the need of inducing the plasma current (and so also avoid instabilities associated to it). We will come back in sect. **9** to this possibility.

3. – The challenge of the heat and particle load

The heat that crosses the separatrix due to conduction losses flows along the magnetic field lines (fig. 4) in a layer a few mm thick and is eventually deposited on the divertor [14]. Since all the heat is localized in a narrow layer, the heat load on the divertor in DEMO can reach values up to $60\,\mathrm{MW/m^2}$, comparable with the heat load at the surface of the Sun!

In addition, the continuous flow of particles impinging on the divertor surface may produce large erosion. A simple estimate of erosion can be done as follows. In ITER with a volume of $800\,\mathrm{m}^3$ and a density of $10^{20}\,\mathrm{m}^{-3}$ there are about 8×10^{22} ions. If the rate at which they are lost through the separatrix is similar to the estimated energy confinement time ($\sim 4\,\mathrm{s}$) the number of particle arriving on the divertor per unit time is $2 \times 10^{22}\,\mathrm{s}^{-1}$. The exposed surface is of the order of $2\,\mathrm{m}^2$. If the probability of extracting an atom from the divertor surface (the so-called sputtering yield) is 10^{-4}, the number of atoms extracted in a year of operation is 3×10^{25} atoms per square meter, or taking into account that solid tungsten has a density of $6.4 \times 10^{28}\,\mathrm{m}^{-3}$, about $0.5\,\mathrm{mm}$ would be eroded every year.

To keep the sputtering yield at low values requires that the temperature in front of the divertor plates must be in the range of few eV. This can be understood as follows.

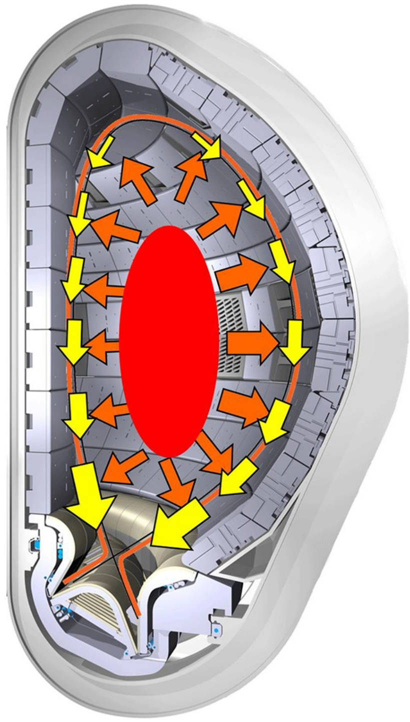

Fig. 4. – Divertor geometry. The orange line is the magnetic separatrix.

Since electrons have a larger mobility than ions they tend to be more easily lost to the divertor plates which become negatively charged with a potential that is a fraction of the electron temperature at the plate divided by the electron charge. This negative potential accelerates the plasma ions towards the plate up to velocities of the order of the local plasma sound velocity. Such a flow of ions is beneficial in opposing the diffusion out of the divertor region of the impurities released from the plates and of the He ashes (thus avoiding plasma contamination and allowing efficient He pumping), however it also increases the probability of impurity extraction. An ion is indeed accelerated to energies proportional to the electron temperature at the plate and the probability of extracting an impurity atom from the plate become significant unless the temperature is low.

The solution to this challenge is made of two recipes. The first recipe is the development of plasma facing components that can withstand high heat and particle fluxes. The solution foreseen for ITER is the so-called tungsten monoblock made by a tungsten dice with a cooling channel made of CuCrZr attached to tungsten through a Cu interlayer. The W-monoblock has been shown to withstand more than 1000 cycles up to $20\,\mathrm{MW/m^2}$. This however would not be enough even for ITER. The second recipe is to produce semi-detached divertor conditions through the formation of a cloud of neutral gas that absorbs energy and momentum of the incoming particles.

In this way it is possible to achieve temperatures in front of the plate of the order

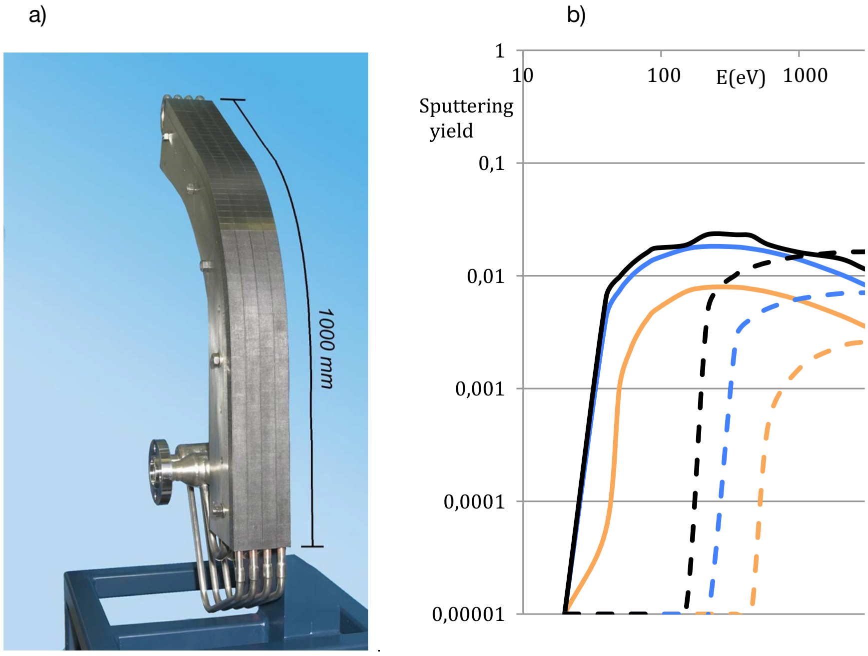

Fig. 5. – Tungsten monoblock mock up (by courtesy of EUROfusion) and physical sputtering yield *vs.* incident ion energy for tungsten (dashed lines) and carbon (continuous lines) under the bombardment by tritium (black), deuterium (blue) and hydrogen (orange).

of few eV, sufficiently low to avoid substantial erosion in stationary conditions (or also under the effect of slow transients) in ITER (see fig. 5(b) where the sputtering yields for C and W under the bombardment of different hydrogen isotopes are plotted using the fits of ref. [15]).

A specific issue is the effect of transient heat loads such as those generated by ELMs and disruptions. Transient loads may drastically reduce the lifetime of plasma facing components (see, *e.g.*, fig. 7 of ref. [14]). Therefore appropriate disruption mitigation systems must be in place and the ELM amplitude must be kept small such that the local deposition is below $1\,\mathrm{MJ/m^2}$ per ELM.

The standard divertor solution is expected to be sufficient to cope with the ITER heat loads. However it is unclear whether it can work also for the much larger heat loads of DEMO. In principle, it can work provided a large fraction of the heating power is radiated before crossing the separatrix. In this case the heat would be exhausted on the large area of the main wall whereas the divertor would continue to play its role for the plasma/impurity density control. However, plasma regimes with high radiation usually exhibit a lower energy confinement time. The development of regimes that simultaneously radiate a large fraction of the heating power from the region inside the separatrix and maintain high confinement is one of the most important research lines for ITER and DEMO.

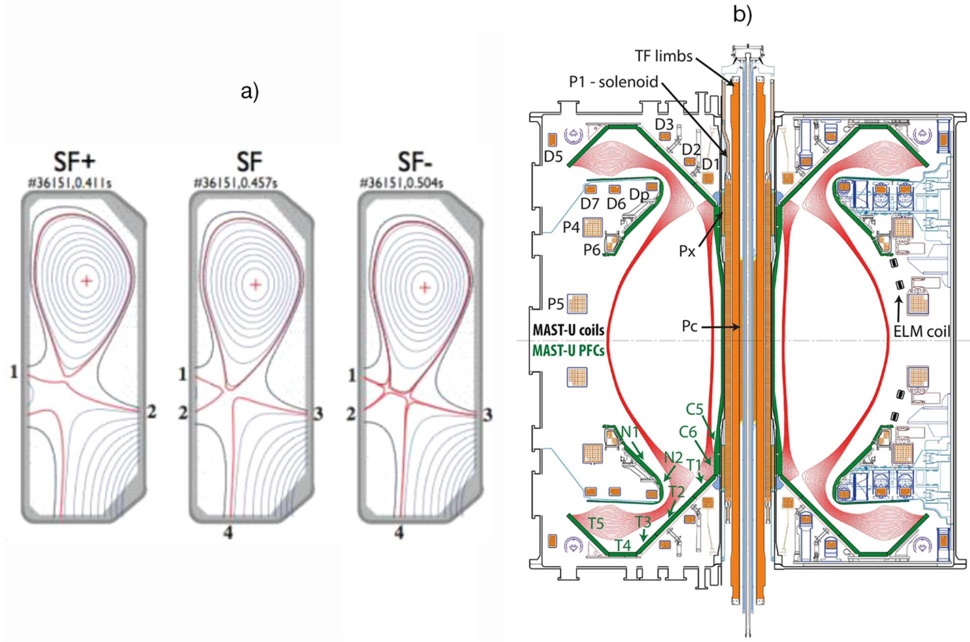

Fig. 6. – (a) Snowflake divertor configuration tested on the TCV tokamak (from ref. [16]). (b) Super-X configuration to be tested on the MAST-U tokamak (from ref. [17]).

What if the predicted DEMO heat load will be too large to be managed with the conventional divertor solution? There is some room of maneuver with the present divertor configuration if the bottom divertor is paired by a divertor at the top or if the magnetic separatrix contact points are slowly swept up and down on the target plate in order to spread the heat load. However, if this will be not enough radically new solutions will be needed. These can be divided in two classes:

– The use of advanced magnetic field configuration with a larger divertor wetted surface. These configurations ("snowflake", super-X, etc. —see fig. 6) are presently tested at the proof of principle level [16, 17].
– The use of liquid metals. In this case the surface exposed to the heat flux is continuously re-formed and its damage is no longer an issue.

The debate about the compatibility of these alternative solutions with the constraints of a reactor is still open. Nevertheless it is clear that the challenge of the exhaust problem deserves a wide investigation. To this goal, the possibility of a Divertor Tokamak Test facility was advocated in the Roadmap [6] and the realization of a dedicated experiment is presently under way (see, *e.g.*, [18]).

4. – Materials that can withstand high neutron fluxes

The use of the DT reaction makes fusion conditions easier to achieve but it has the drawback that 80% of the fusion power is released in the form of 14 MeV neutrons. Part of the fusion-generated neutrons does not react with the lithium in the blanket to produce tritium but is absorbed by the structural materials of the reactor. This produces a degradation of structural properties and activation of wall materials. This degradation is mostly localized in the first few tens of centimetres (from the plasma exposed surface) of the plasma facing components. The vacuum vessel receives very low levels of neutron irradiation and it maintains its structural properties throughout the life of the reactor. On the contrary, the plasma facing components will need to be replaced every 3-4 years.

The damage suffered by structural materials can be quantified in terms of the displacements per atom (dpa). The nucleus in the structural material lattice that has absorbed the neutron energy (primary knock-on atom) releases its energy by displacing the surrounding atoms and producing point defects and dislocations. This effect is measured by the average number of displacements per atoms (dpa) in the lattice. Although deterioration of structural properties due to neutron irradiation is encountered also in fission reactors, the high energy of fusion neutrons (14 MeV compared with around 2 MeV for neutrons produced in fission reactors) induces a substantial amount of reactions forming H or He that accumulates in the lattice. These two effects produce embrittlement, swelling and irradiation creep and become important beyond the level of ~ 10 dpa [19,20]. As a result the temperature window for operation is limited between 350 °C and 550 °C. Typical numbers for radiation damage are 100–150 dpa in a fusion reactor, 30–70 dpa in DEMO and 2 dpa in ITER. Thus, there is no problem with material degradation in ITER. However, appropriate materials must be developed and qualified for fusion reactor application to avoid frequent replacement of the plasma facing components and the blanket. For DEMO, it is possible to consider an initial exploitation with the presently qualified materials, but the second phase will require significant advances in material performance.

In view of the high-energy neutrons that characterize fusion, such a qualification will require a dedicated facility, the International Fusion Material Irradiation Facility (IFMIF). IFMIF will produce a neutron spectrum similar to that of a fusion reactor through various stripping reactions between two 40 MeV/125 mA deuteron beams and a liquid lithium target. The engineering validation and engineering design activity are presently being finalized within the framework of a collaboration between the EU and Japan [21].

Activation is the second issue of neutron irradiation. Since the fusion reaction does not produce radioactive materials (the primary source of waste for fission), the issue of activation is limited to the structural materials of the plasma facing components. Among the structural materials so far investigated, reduced activation ferritic-martensitic steels (RAFM, such as EUROFER [22]), appears to be the near-term solution. RAFM steels differ from austenitic steels as molybdenum, nickel and niobium are replaced by tungsten, tantalum, vanadium and/or titanium that have a better behaviour under neutron irradiation. These alloys achieve a sufficiently low level of radioactivity in a sufficiently short time (say 100 years after the end of the reactor operation) in such a way that all the reactor materials can be easily recycled in a new reactor. Simple

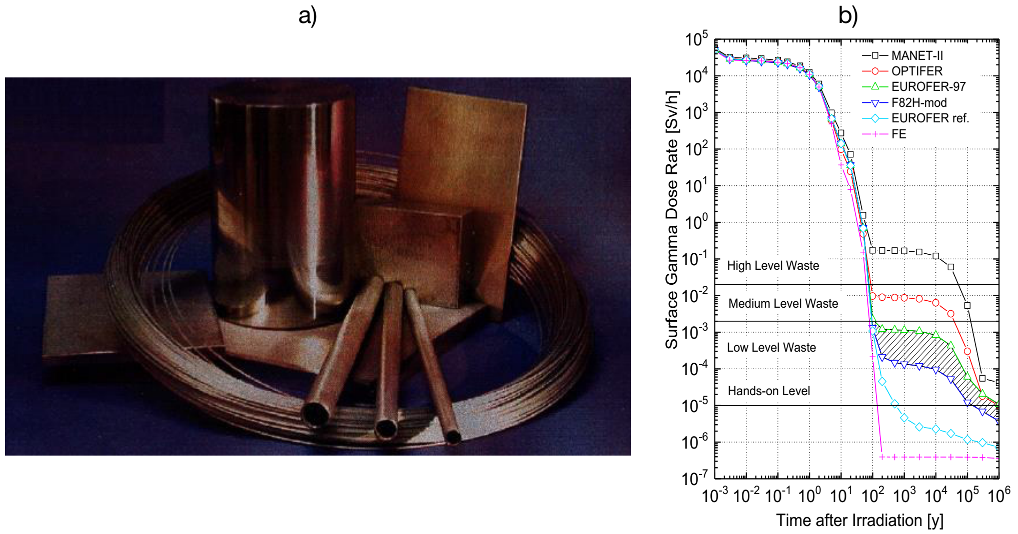

Fig. 7. – (a) Picture of EUROFER components (by courtesy of EUROfusion) and; (b) contact dose rate *vs.* time for different steels from ref. [23].

recycling techniques are expected to apply below contact dose rates $\leq 2\,\mathrm{mSv/h}$, whereas for a contact dose rate $\leq 20\,\mathrm{mSv/h}$ recycling is still considered possible although through more complex remote handling systems. For comparison, the hands-on limit (*i.e.*, the contact dose rate that allows a maximum dose of $20\,\mathrm{mSv/y}$ for a radiation exposed worker) corresponds to $10\,\mu\mathrm{Sv/h}$.

In its ideal composition, EUROFER (Cr 9% W 1.1% Mn 0.4% V 0.2% C 0.11% Ta % Si 0.05% N 0.03% Ti 0.01% and Fe for the rest) would achieve a level of radiation below $2\,\mathrm{mSv/h}$ in less than 100 years and the hands-on limit in about 400 years. The EUROFER produced today (the so-called EUROFER97, see fig. 7a) still contains some impurities but can already reach the simple recycling limit in ~ 100 years after shut down, see ref. [23] and fig. 7b. Thus, for fusion power no geological repository will be necessary.

The development of advanced steels that would allow a wider operating temperature window is an active area of research. Oxide dispersion strengthened steel is expected to allow operation up to $650\,^{\circ}\mathrm{C}$. A review of the present status of materials for reactor applications can be found in ref. [24].

5. – Demonstration of tritium self-sufficiency

Tritium does not exist in nature. It must be produced inside the reactor. A 1.5 GWe reactor will consume about 0.5 kg of tritium per day and will need to produce the same amount every day. A tritium breeding-ratio around 1.1–1.15, is the target for the self-sustainability of a reactor.

Tritium is produced through the reaction between neutron and lithium

$$\mathrm{n} + {}^{6}\mathrm{Li} \rightarrow {}^{4}\mathrm{He}\ (2.1\,\mathrm{MeV}) + \mathrm{T}\ (2.7\,\mathrm{MeV}),$$

$$\mathrm{n} + {}^{7}\mathrm{Li} \rightarrow {}^{4}\mathrm{He} + \mathrm{T} + \mathrm{n} - 2.5\,\mathrm{MeV}.$$

The reaction involving ^{6}Li is exoergic and provides an additional contribution to the thermal power of the reactor. Its cross section at low energy is large (> 100 b at 1 eV). The reaction involving ^{7}Li is endoergic. It has a threshold at 2.5 MeV so only high-energy neutrons can be used. Its maximum cross section is below 1b. Therefore the lithium of a reactor must be enriched in ^{6}Li. The tritium production using these two reactions has been tested at low 14 MeV neutron fluence [25]. These show good agreement between the calculated and experimental production of tritium.

Some of the neutrons are lost due to various processes (absorption, streaming through the ports, etc.). Thus, neutron multipliers must be used. The main candidates are beryllium and lead. The (n, 2n) reaction of Be has a lower cross section (~ 0.5 b) but also a lower neutron energy threshold (1.7 MeV). In the case of Pb the cross section is larger (~ 2 b) but the neutron energy threshold is at 7.4 MeV. The use of Be as a multiplier is made in connection to the solid breeder concept in which ceramic compounds (Li_4SiO_4, Li_2TiO_3 or others) are produced in the form of pebbles. The breeder is separated from the Be multiplier and the two are cooled by Helium (water is avoided due to the reactivity with Be). The use of Pb is made in connection with the liquid breeder concept in which the breeder and the multiplier form a liquid eutectic compound ($Li_{17}Pb_{83}$) that flows at low velocity and is cooled by either He or water. There are advantages and disadvantages in both concepts and one of the ITER goals is to test blanket prototypes to verify the modelling assumptions [26].

6. – The intrinsic safety features of fusion

Fusion has a feature that makes it attractive within the nuclear technologies: its intrinsic safety. No chain reactions take place in a fusion power plant. The main difference between fission and fusion power plants is that a fission power plant is like a pile (the Fermi pile!): all the energy that will be released is stored in the fuel initially placed in the reactor. A recovery action following a malfunctioning during operations may need to cope with the release of a large amount of power. A fusion plant is like a normal gas boiler, in order to run it has to be continuously fuelled. If something goes wrong it is sufficient to stop fuel from entering the reaction chamber and the reactor stops. In both cases there is always a residual decay heat that has to be exhausted to avoid melting of the confinement structures (this was indeed the problem with the Fukushima accident). However, in the case of fusion the decay heat can be exhausted using only passive conduction. Indeed the incidental studies done so far show that the temperature of the in-vessel components in a fusion power plant always remains below the melting temperature [27].

7. – The integration of advanced technologies

Fusion power plants require the integration of different technologies. A large number of these technologies have been developed in connection with the ITER R&D activities such as those related with large superconducting magnets and heating and current drive systems. DEMO will have to integrate the technologies for tritium breeding (tested at the level of proof-of-principle in ITER) and for the production of electricity through a balance-of-plant (BoP) system. The integration of different technologies motivates a focused effort on the solutions that can be really implemented in a reactor.

The technology of low-temperature superconducting magnets is today well established. The challenges of the ITER magnets [28] have been addressed within a dedicated R&D programme during the Engineering Design Activity [29]. The design and construction of the magnet system for DEMO requires limited extrapolations of the ITER design and ensures that a technical solution already exists while fusion-relevant high-temperature superconductors are developed in parallel. R&D activity is ongoing on more advanced low-temperature superconducting cables in order to reduce degradation under cyclic operation and cost. The integration of the magnet systems into DEMO is expected to largely build on the ITER experience.

ITER will test three auxiliary heating and current drive systems: neutral beam injection, electron cyclotron heating and current drive and ion cyclotron heating. Their use in ITER and DEMO poses specific technical challenges such as the development of high-power, continuous source (gyrotrons) for millimetre wave radiation (170 GHz) for electron cyclotron heating and current drive and large negative ion sources and accelerator systems for negative ion injection ($1\,\mathrm{MeV}/200\,\mathrm{A\,m^{-2}}$). The choice of the system will have to be made mostly on the basis of the impact on tritium breeding (the opening in the vacuum vessel for the power injection are lost to the breeding blanket), on the degree of reliability and on the demonstration of sufficiently high efficiency to avoid large amounts of re-circulating power (see ref. [30] for a recent review and references therein).

A large impact on the design will come from the requirements of minimizing the plant down time through an effective remote maintenance system. The scheme presently under development foresees the use of large vertical sectors (vertical maintenance scheme) [31].

The integration of the DEMO components with the BoP has been investigated in the last few years. The choice of the BoP has a number of consequences on the choice of blanket coolant and materials [32].

8. – Electricity at low cost from fusion

The cost of fusion electricity has been the subject of several studies using the standard levelised cost of electricity approach by the IEA that includes the future expenditure (capital, operation and maintenance, replacements, fuel and decommissioning charges) all discounted to present day. With reasonable assumptions on the technology learning factor (based on the present experience with several energy technologies), on the plant availability (75%) and lifetime (40 years) the cost of the fusion kWh is estimated to be between 5 c€/kWh and 10 c€/kWh [27] in line with present market values. Most of the cost depends on the capital investment whereas fuel costs have a negligible impact.

DEMO is expected to be a relatively small extrapolation with respect to ITER. Its major radius should be at most 50% larger than ITER if the present physics and technological basis will be confirmed by future R&D. Therefore extrapolations to DEMO of the ITER experience in many areas can be made with confidence.

Nevertheless the experience with the ITER costs and with the costs of several nuclear power plants under construction shows that particular attention has to be devoted to this challenge. DEMO is not expected to produce electricity at a competitive price but should demonstrate that the capital costs can be contained to a level that makes fusion a competitive energy source in the long term.

9. – Stellarators

Achieving steady-state operation in a tokamak is made difficult by the need of operational regimes of high plasma pressure that are more prone to plasma instabilities and require a sophisticated control system. A radical solution is to abandon the tokamak line and consider non-axisymmetric configurations such as the stellarator [33]. In these configurations the rotational transform that is needed to confine the plasma is not produced by inducing a net plasma current but through a set of external coils.

A non-axisymmetric configuration has always regions of stochastic magnetic field that act as a thermal short circuit and tend to degrade energy confinement. However, the configuration can be optimized to reduce the size of these regions and their impact on global confinement.

The difficulty of manufacturing external coils with a complex structure is balanced by the avoidance of the plasma current. In addition, the absence of a plasma current also reduces one of the major free-energy source driving plasma disruptions.

Stellarator research has made significant progress. The energy confinement is well characterized and the exploitation of the new W7X device in Germany has started.

If the excessive occurrence of disruptions will make the tokamak line not suited for commercial reactor applications, the stellarator line will be a potentially valid alternative that will make full use of the R&D achievements obtained during the construction and operation of DEMO.

10. – Concluding remarks

Fusion has the potential to become a major source of electricity. The research carried out in the last 50 years allows today to routinely produce plasmas at reactor relevant density and temperature. Substantial progresses have been also made in addressing the key technological challenges.

The demonstration of plasma operations with alpha particles being the dominant heating mechanism will be achieved on ITER. On the basis of the present theoretical and experimental evidence a fusion gain $Q = 10$ is expected. Methods to avoid, prevent or mitigate plasma disruptions and to control benign plasma instabilities have been developed and must now be tested in ITER. ITER will also show the viability of fully steady-state operations for reactor application.

The challenge of coping with the heat exhaust will be addressed in ITER through the use of high-heat flux components based on the tungsten monoblock technology and the use of partially detached divertor operation. In order to mitigate the risk that this solution cannot be extrapolated to DEMO, schemes based on the use of advanced divertor configuration and liquid metals as plasma facing materials are being investigated.

The qualification of structural materials that can withstand the intense neutron flux of a fusion reactor and have benign activation properties needs to be further pursued. Possible materials (such as EUROFER) have already been produced. They could be used with limited extrapolations at the beginning of DEMO operation since the level of nuclear damage is expected to be below 20 dpa for the first phase of the DEMO operation. For the second phase of DEMO operation new materials need to be qualified. This requires a specific facility (IFMIF) with a neutron energy spectrum that simulates that of a fusion reactor.

Efficient tritium breeding is mandatory in a fusion power plant since about 0.5 kg of tritium are burned every day and an equal amount has to be produced in the blanket and re-circulated in the reactor. Blanket technologies based either on the liquid eutectic LiPb compound or the solid ceramic breeder with Be as neutron multiplier are under development and need to be integrated into a coherent design that accounts for the limited operating temperature window of the presently available materials, the need of minimizing tritium permeation outside the fuel cycle and the requirements of the balance of plant system to produce a net electrical power output.

ITER will demonstrate many of the critical technologies for a fusion power plant: magnets based on the use of low-temperature superconductivity, construction of large scale vacuum vessel with strict tolerances, actively cooled plasma facing components under high heat loads, high efficiency auxiliary heating system, fuel cycle technologies capable of managing large tritium inventories. DEMO will largely benefit of this experience but will have to ensure reliable breeding blanket technologies and efficient balance of plant systems at sufficiently low cost.

Stellarator is a promising alternative to the tokamak line due to its intrinsically steady-state operation and lack of disruptions. The ongoing R&D will have to develop into a burning plasma stellarator experiment in due time to demonstrate its viability as a fusion power plant.

The National Ignition Facility in operation at Livermore since 2009 will demonstrate the potential of inertial confinement fusion. For the first time fusion energy larger than the fuel energy at the burn time has been achieved with about 50% of the reactions due to self-heating [34]. Although the process to produce fusion energy is totally different from that of magnetic confinement, many of the technological challenges (such as neutron resistant material and tritium breeding) are common to magnetic and inertial confinement.

A successful R&D focused on the program priorities and the first operational experience on ITER may allow the construction of a DEMO reactor and the demonstration of fusion electricity around the middle of this century.

REFERENCES

[1] *A European strategic energy technology plan (SET-plan) - Towards a low carbon future* COM(2007)723.
[2] Vienna Standard Mean Ocean Water, *Reference Sheet for International Measurement Standards* (IAEA Vienna) Dec 2006, `https://nucleus.iaea.org/rpst/documents/VSMOW_SLAP.pdf`.
[3] Rosman K. J. R. and Taylor P. D. P. (Editors) *Pure Appl. Chem.*, **70** (1998) 217.
[4] Atzeni S. and Meyer-ter-Vehn J., *The Physics of Inertial Fusion* (Oxford University Press) 2004.
[5] Keilhacker M. (on behalf of the JET Team), *Nucl. Fusion*, **39** (1999) 209.
[6] Romanelli F. *et al.*, *Fusion electricity - A Roadmap to the realization of fusion energy* ISBN: 978-3-00-040720-8, `https://www.euro-fusion.org/wpcms/wp-content/uploads/2013/01/JG12.356-web.pdf`.
[7] `https://www.euro-fusion.org/eurofusion/roadmap/`.
[8] Wesson J., *Tokamaks*, third edition (Clarendon Press, Oxford) 2004.
[9] ITER Physics Expert Groups on Confinement and Transport and Confinement Modelling and Database, ITER Physics Basis Editors and ITER EDA, *Nucl. Fusion*, **39** (1999) 2175.
[10] Bosch H. S. and Hale G. M., *Nucl. Fusion*, **32** (1992) 611.
[11] Hender T. C. *et al.*, *Nucl. Fusion*, **47** (2007) S128.
[12] Fasoli A. *et al.*, *Nucl. Fusion*, **47** (2007) S264.
[13] Jacquinot J. *et al.*, *Nucl. Fusion*, **38** (1998) 2495.
[14] Loarte A. *et al.*, *Nucl. Fusion*, **47** (2007) S203.
[15] Matsunami N. *et al.*, IPP-AM-32 report Nagoya (1983).
[16] Piras F. *et al.*, *Plasma Phys. Control. Fusion*, **51** (2009) 055009.
[17] Morris W. *et al.*, *IEEE Trans. Plasma Sci.*, **46** (2018) 1217.
[18] Albanese R. *et al.*, *Fus. Eng. Des.*, **146** (2018) 194.
[19] Baluc N. *et al.*, *Nucl. Fusion*, **47** (2007) S696.
[20] Baluc N., *Phys. Scr. T*, **138** (2009) 014004.
[21] Knaster J. *et al.*, *Nucl. Fusion*, **57** (2017) 102016.
[22] van der Schaaf B. *et al.*, *J. Nucl. Mater.*, **283-287** (2000) 52.
[23] Lindau R. *et al.*, *Fus. Eng. Des.*, **75-79** (2005) 95.
[24] Stork D. *et al.*, *J. Nucl. Mater.*, **455** (2014) 277.
[25] Batistoni P. *et al.*, *Fus. Eng. Des.*, **82** (2007) 2095.
[26] Federici G. *et al.*, *Fus. Eng. Des.*, **141** (2019) 30.
[27] Maisonnier D. *et al.*, *Nucl. Fusion*, **47** (2007) 1524.
[28] Mitchell N., *Fus. Eng. Des.*, **66** (2003) 971.
[29] Mitchell N. *et al.*, *Supercond. Sci. Technol.*, **20** (2007) 25.
[30] Franke T. *et al.*, *IEEE Trans. Plasma Sci.*, **46** (2018) 1633.
[31] Keep J. *et al.*, *Fus. Eng. Des.*, **124** (2017) 420.
[32] Barucca L., *Fus. Eng. Des.*, **136** (2018) 1557.
[33] Klinger T. *et al.*, *Nucl. Fusion*, **59** (2019) 112004.
[34] Hurricane O. A. *et al.*, *Nature*, **506** (2014) 343.