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Fusion energy: technological challenges

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Summary. — This paper presents an overview of the main technological challenges of magnetic confinement fusion. Many of the challenges are being addressed in the context of the ITER construction and exploitation. Specifically, the demonstration of high-fusion gain regimes of operation will also provide a test of the technological solutions presently foreseen for 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

The world is vigorously pursuing a net zero CO2 emission strategy [1]. Fusion can play an important role in the future mix of energy technologies since it has a number of advantages:

- It is virtually unlimited.
- It does not produce greenhouse gases.
- It is intrinsically safe.
- 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 no geological repository is necessary.

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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%
$\rightarrow {}^3He\ (0.82\,MeV) + n\ (2.45\,MeV)$	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%[(a)]
$p + {}^6Li \rightarrow {}^4He\ (1.7\,MeV) + {}^3He\ (2.3\,MeV)$	
$p + {}^7Li \rightarrow 2{}^4He + 17.3\,MeV$	20%
$\rightarrow {}^7Be + n - 1.6\,MeV$	80%
$D + {}^6Li \rightarrow 2{}^4He + 22.4\,MeV$	
$p + {}^{11}B \rightarrow 3{}^4He + 8.7\,MeV$	

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

Several fusion reactions can in principle be exploited (table I). However, the reaction with the largest cross section is the reaction between deuterium and tritium that produces a 3.5 MeV alpha particle and a 14 MeV neutron. The amount of deuterium in sea-water (about 35 mg per litre [2, 3]) is virtually unlimited (making fusion with deuterium alone would supply the Earth, at the present consumption rate, for several billions years!). Tritium instead is radioactive and undergoes beta decay with a half-life of 12.3 years. Therefore, it has to be produced inside the reactor. This is accomplished by covering the inner part of the reaction chamber with a lithium blanket. 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.

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^3He and DD reactions. In order to fuse D and T their mutual Coulomb repulsion must be overcome. This can be achieved by heating the DT gas to sufficiently high temperature (~ 20 keV or 200 M°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 magnetic or inertial confinement [4] can be employed. We will consider here only magnetic confinement. In this case we can take advantage of the charged nature of the plasma constituents to confine them using intense magnetic fields (~ 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

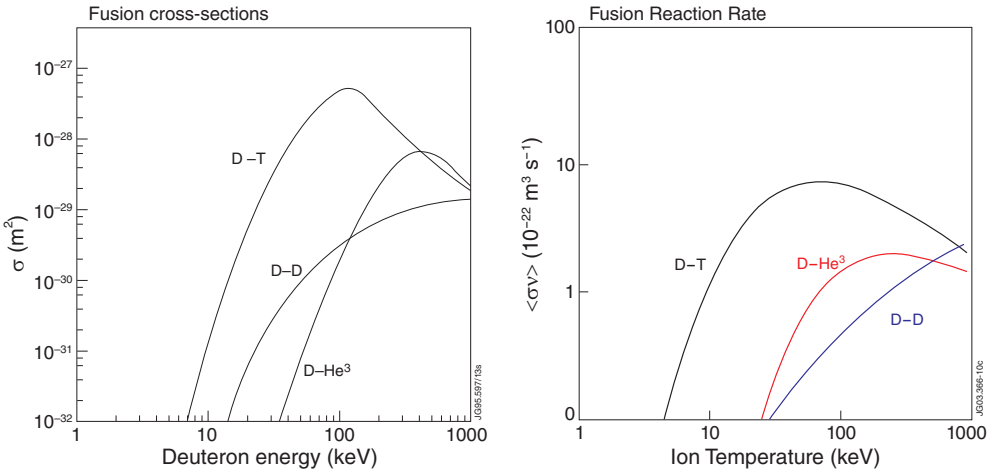

Fig. 1. – Fusion cross sections and fusion reactivity for the three main fusion reactions. The cross section is plotted as a function of the energy of the deuteron impinging on the target at rest.

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 ($\sim 5\,\mathrm{T}$) and toroidal geometry enables plasma confinement.

It is important to stress that with both magnetic and inertial confinement 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. Moreover, the National Ignition Facility, the largest inertial confinement fusion facility, has recently reported the production of 1.3 MJ of fusion energy which is about 70% of the energy delivered to the target [6]. 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 [7] (an update has been recently produced by EUROfusion [8]) 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.

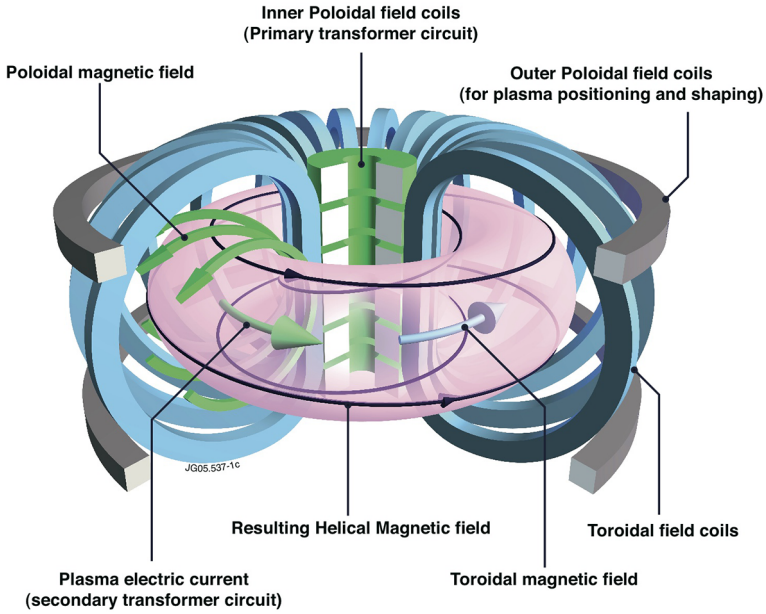

Fig. 2. – Magnetic confinement geometry illustrated for the tokamak configuration, see text (courtesy EUROfusion).

– 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* [9] and schematically shown in fig. 2. The plasma is characterized by a doughnut shape (*torus*) with major radius R_0 and minor radius a. The magnetic field is the superposition of a toroidal component B_ϕ (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 toroidal and 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 inhomogeneity, 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.

The safety factor is constant on a magnetic surface. For the sake of discussing the technological challenges we can use a simple approximation to the safety factor at the plasma edge [9]

$$q_{\rm edge} = 2\pi B_\phi a^2 S/(R_0 \mu_0 I_p), \tag{1}$$

where I_p is the plasma current and S is a numerical factor that depends on the plasma shape. In order to have stable plasma operations [9] a necessary condition is $q_{\rm edge} \geq 3$, which produces an upper bound to the plasma current that can circulate in a tokamak

$$I_{p,\max} = 2\pi B_\phi a^2 S/(3\mu_0 R_0). \tag{2}$$

Since, as we will see later, the plasma current is the main control parameter to achieve high fusion gain, the limit expressed by eq. (1) has to be taken into account to design a tokamak reactor. It is apparent that at fixed plasma shape and aspect ratio R_0/a, the maximum plasma current is proportional to $B_\phi a$. Thus, $I_{p,\max}$ can be increased either by increasing the reactor dimensions or by increasing the toroidal field.

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 the technological challenges of magnetic confinement although a number of challenges are common also to inertial confinement. In another paper of this school [10] the physical challenges will be reviewed in detail.

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). This challenge is briefly reviewed in this section in order to make contact with the technological challenges. A detailed discussion can be found in ref. [10]. 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). Specifically, the power generated by fusion reactions is given by

$$P_{\rm fus} = 17.6\,{\rm MeV}\, n_{\rm D}\, n_{\rm T} \langle \sigma v \rangle V, \tag{3}$$

where $n_{\rm D}$ is the particle density of the deuterium nuclei (number of deuterium nuclei per unit volume), $n_{\rm T}$ is the particle density of the tritium nuclei, $\langle \sigma v \rangle$ is the Maxwellian fusion reactivity and V is the plasma volume. Since the 14 MeV neutrons carry four times more energy than the 3.5 MeV alpha particles, 20% of the power given by eq. (3) is in the form of alpha particles and 80% in the form of neutrons. The fusion power produced in the form of neutrons in ITER is 400 MW and 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 $Q = 500\,{\rm MW}/50\,{\rm MW} = 10$. 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 [11], whereas the fusion power increases as R^3. Therefore the solution that has been pursued has been to make the machine size larger.

The power $P_{\rm cond}$ 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

$$P_{\rm cond} = W/\tau_E, \tag{4}$$

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 balance equation in stationary conditions dictates that the fusion power released in the form of alpha particles (3.5 MeV/17.6 MeV $\sim$ 20% of the total) plus the power $P_{\rm aux}$ injected from external sources must be equal to the power lost by the plasma. Neglecting the power lost by radiation we have

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

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

For a pure deuterium and tritium plasma ($n_{\rm D} = n_{\rm T} = n_e/2$) with equal electron and ion temperatures, the ignition condition can be written in terms of the so-called triple product([1])

$$n_e \tau_E T \geq 3.42 \times 10^{21}\,{\rm m}^{-3}{\rm s\,keV}. \tag{6}$$

A careful analysis of the experimental data has produced the following scaling law for the energy confinement time in H-mode [11]

$$\tau_{{\rm ITER98}(y,2)}({\rm s}) = \tag{7}$$

$$0.0562\, I_p({\rm MA})^{0.93}\, B({\rm T})^{0.15}\, P({\rm MW})^{-0.69}\, n(10^{19}\,{\rm m}^{-3})^{0.41}\, M^{0.19}\, R_0({\rm m})^{1.97} \epsilon^{0.58} \kappa^{0.78},$$

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, with an estimated value of the triple product (assuming $n = 10^{20}\,{\rm m}^{-3}$ and $T = 20\,{\rm keV}$) $n_e \tau_E T \sim 8.6 \times 10^{21}\,{\rm m}^{-3}{\rm s\,keV}$.

We refer to ref. [10] for a discussion of the ignition condition. In the context of the technological challenges we want to stress that upon combining the equations above it is possible to obtain an expression of the triple product in terms of the reactor parameters

$$n_e \tau_E T \propto n^{0.8} R^{0.7} P^{-0.4} {I_p}^{2.2}.$$

This expression shows that the triple product is mainly determined by the plasma current. Since, as noted above, the maximum plasma current is proportional to the product $B_\phi a$, high fusion gain conditions can be obtained either by small-dimensions high magnetic field devices or large-dimensions low magnetic field devices. The use of high magnetic fields leads to a more compact reactor design, but this path has been so far limited by the availability of superconducting materials capable of operating above 12 T for typical

([1]) In deriving eq. (6) we have approximated the fusion reactivity as $\langle \sigma v \rangle \sim 10^{-24} T\,({\rm keV})^2 {\rm m}^3/{\rm s}$, a good approximation in the range 10–20 keV.

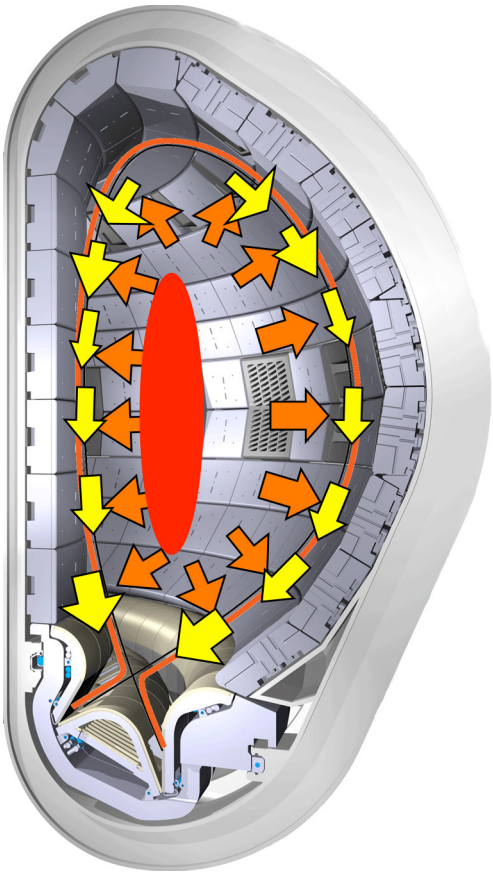

Fig. 3. – Divertor geometry. The orange line is the magnetic separatrix.

values of the current density in the magnet [12]. Recent developments in this field [13] have shown that coils capable of operating up to 20 T are indeed possible using high-temperature superconductors operated at liquid He temperatures.

Although the results of ref. [13] are indeed remarkable, it should be stressed that the design of compact fusion reactors must take into account all the constraints arising from the increase in B_ϕ.

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. 3) 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 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. 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 becomes 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 the energy and momentum of the incoming particles.

In this way it is possible to achieve temperatures in front of the plate of the order of few eV, sufficiently low to avoid substantial erosion in stationary conditions (or also under the effect of slow transients) in ITER (see fig. 4 (right)) where the sputtering yields for C and W under the bombardment of different hydrogen isotopes are plotted using the fits of ref. [14]).

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. [15]). 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

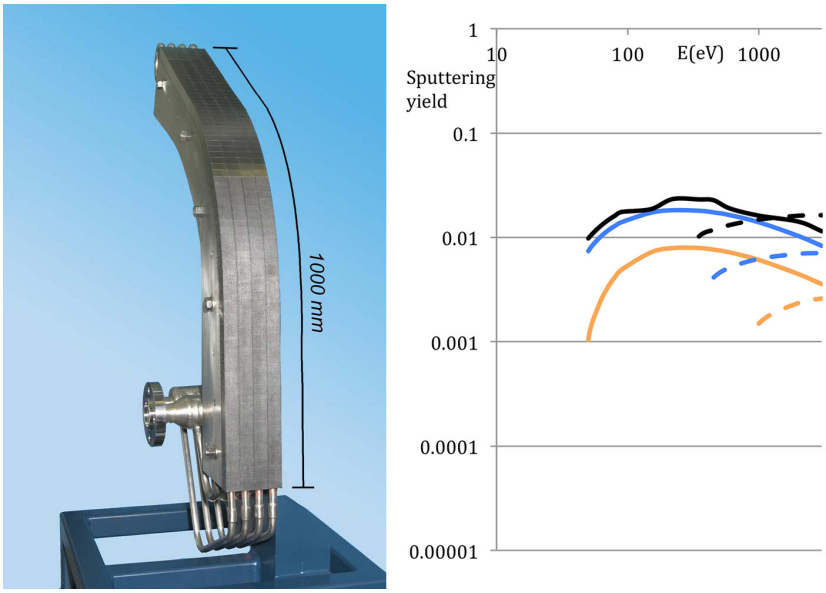

Fig. 4. – Tungsten monoblock mock up (courtesy EUROfusion) and physical sputtering yield *vs.* incident ion energy for tungsten (dashed lines) and carbon (continuous lines) under the bombardment by tritium (black), deuterium (blue) and hydrogen (orange).

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.

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.) are presently tested at the proof of principle level [16, 17] (see fig. 5).

– 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

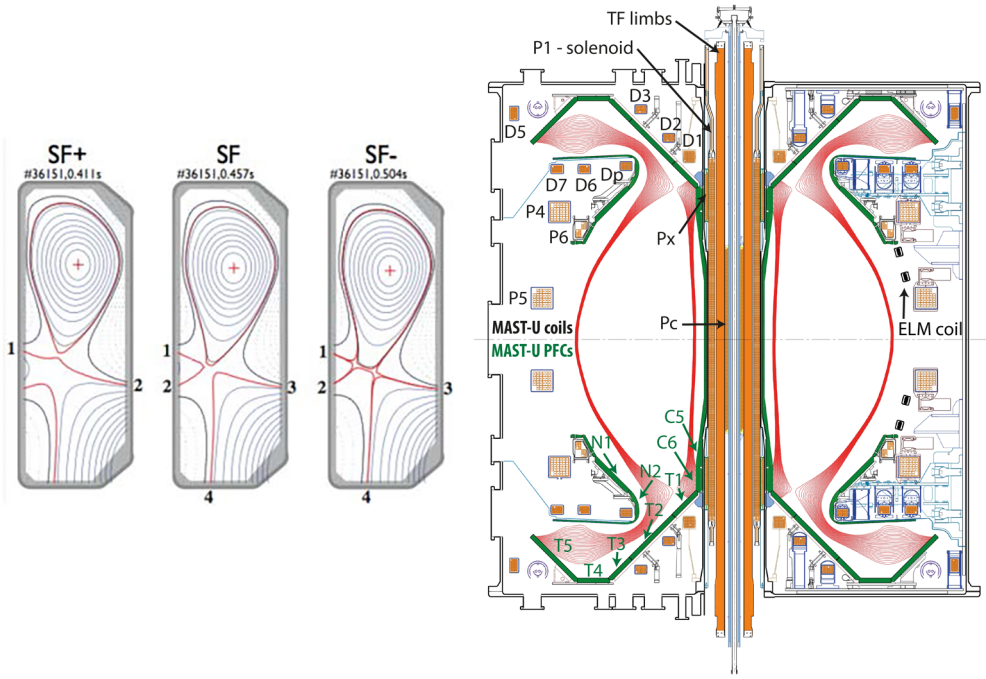

Fig. 5. – (a) Snowflake divertor configuration tested on the TCV tokamak (from ref. [16]). (b) Super-X configuration to be tested on the MAST-U tokamak (from ref. [17]).

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 [18].

DTT is a superconducting tokamak capable of producing plasmas with a current up to 5.5 MA and heating power up to 45 MW for long pulses (100 s) that will investigate innovative solutions for the divertor in various plasma configurations. It is presently being built at the Frascati ENEA laboratory([2]).

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

([2]) `https://www.dtt-project.it/`.

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 the 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 recycling techniques are expected to apply below contact dose rates ≤ 2 mSv/h, whereas for a contact dose rate ≤ 20 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 mSv/y for a radiation exposed worker) corresponds to $10\,\mu$ 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

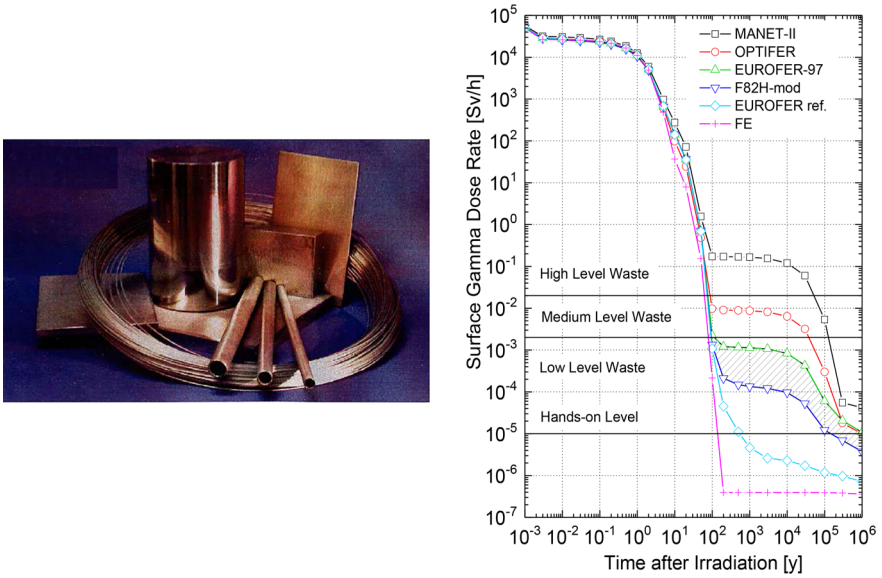

Fig. 6. – (a) Picture of EUROFER components (courtesy EUROfusion) and (b) contact dose rate *vs.* time for different steels from ref. [23].

2 mSv/h in less than 100 years and the hands-on limit in about 400 years (see fig. 6). The EUROFER produced today (the so-called EUROFER97) still contains some impurities but can already reach the simple recycling limit in $\sim$ 100 years after shut down [23]. 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 °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 1 b. 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, *i.e.* a system of heat exchangers and conventional turbines to convert into electricity the heat generated by the fusion reactions. 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. In parallel, the development of high temperature superconducting coils [13] will demonstrate the feasibility of more compact fusion reactors.

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 ($1\,\mathrm{MeV}/200\,\mathrm{Am}^{-2}$). 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. – 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.

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 and a dedicated DTT facility is under construction.

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 a 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 an efficient balance of plant systems at sufficiently low cost.

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] https://www.iea.org/reports/net-zero-by-2050.
[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] https://www.llnl.gov/news/national-ignition-facility-experiment-puts-researchers-threshold-fusion-ignition.
[7] 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.
[8] https://www.euro-fusion.org/eurofusion/roadmap/.
[9] Wesson J., *Tokamaks*, third edition (Clarendon Press, Oxford) 2004.
[10] Ongena J., these proceedings, p. 143.
[11] "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.
[12] Bruzzone P. *et al.*, *Nucl. Fusion*, **58** (2018) 103001.
[13] https://news.mit.edu/2021/MIT-CFS-major-advance-toward-fusion-energy-0908.
[14] Matsunami N. *et al.*, IPP-AM-32 report Nagoya (1983).
[15] Loarte A. *et al.*, *Nucl. Fusion*, **47** (2007) S203.
[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. Design*, **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. Design*, **75-79** (2005) 95.
[24] Stork D. *et al.*, *J. Nucl. Mater.*, **455** (2014) 277.
[25] Batistoni P. *et al.*, *Fusion Eng. Des.*, **82** (2007) 2095.
[26] Federici G. *et al.*, *Fusion Eng. Des.*, **141** (2019) 30.
[27] Maisonnier D. *et al.*, *Nucl. Fusion*, **47** (2007) 1524.

[28] Mitchell N., *Fusion 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.*, *Fusion Eng. Des.*, **124** (2017) 420.
[32] Barucca L., *Fusion Eng. Des.*, **136** (2018) 1557.