THE BIG STEP FROM ITER TO DEMO

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ABSTRACT

Extrapolation of the knowledge base towards a future fusion power reactor is discussed. Although fusion research has achieved important milestones since the start and continues achieving successes, we show that there are still important challenges that have to be addressed before the construction and operation of an economical fusion power reactor. Both physics and technological questions have to be solved. ITER is a significant step that will lead to major progress; for a DEMO reactor however, there will still be outstanding physics and engineering questions that require further R&D. This paper introduces some of the main topics to illustrate the challenges that lie in front of us.

I. INTRODUCTION

The decision to build ITER has been a very important step towards the realization of a fusion energy source. However, as we will show in the paper, there are still major challenges beyond ITER that must be resolved. DEMO, should be a practical demonstration of electricity generation on a power-plant scale that satisfies various socio-economic goals; it will include a closed tritium fuel cycle, and demonstrate a high level of safety and low environmental impact. Such a DEMO device will be a major milestone towards a fusion energy source that can economically compete with other energy sources. DEMO programmes are different in different parts of the world, although there is the common plan to try to have an operational DEMO device around the middle of this century.

DEMO is currently based on the tokamak, as this is the most advanced fusion concept to date, and plasma parameters approaching those of a reactor are foreseen in ITER. Reactor studies are also being developed for Helical Devices (see e.g. [1-4]). However, a decision on a next step stellarator/helical device can only take place when the main results of the current large helical devices in operation or construction have been obtained. The largest helical device currently in operation is LHD (Large Helical Device, in the National Institute for Fusion Science (NIFS), close to Nagoya, Japan). The construction of the largest stellarator in the world Wendelstein 7-X (Max-Planck Institute, Greifswald, Germany) is nearly finished with first operations foreseen for beginning 2015. In this paper, we will therefore restrict ourselves to the discussion of a tokamak fusion reactor.

The European Fusion Development Agreement (EFDA) has released recently (November 2012) a roadmap for the realization of fusion electricity to the grid by 2050 [5]. This roadmap covers three periods: (i) the upcoming European Research Framework Programme, Horizon 2020, (ii) the years 2021-2030 and (iii) the period 2031-2050.

ITER is the key facility of the roadmap as it is expected to achieve most of the important milestones on the path to fusion power. The vast majority of resources proposed for Horizon 2020 are dedicated to ITER and its accompanying experiments. The second period is focused on maximizing ITER exploitation and on preparing the construction of DEMO. Building and operating DEMO is the subject of the last roadmap phase (time horizon about 2050).

To lead a coordinated effort in the EU (building on efforts done in the past) towards DEMO, the Power Plant Physics and Technology Department (PPP&T) has been established under EFDA in 2011 [6].

The aims of the DEMO studies in Europe are:

- to quantify key physics and technology prerequisites for DEMO;
- to identify the most urgent technical issues that need to be solved in physics and technology;
- to plan and implement supporting physics and technology R&D.

Two DEMO design options are currently being investigated by PPP&T. (See Table I for main characteristics):

- DEMO Model 1: A "conservative baseline design" that could be delivered in the short to medium term, based on the expected performance of ITER with reasonable improvements in science and technology i.e. a large, modest power density, long-pulse inductively supported plasma in a conventional plasma scenario.

- DEMO Model 2: an "optimistic design" based upon more advanced assumptions which are at the upper limit of what may be achieved, leading to a steady state plasma scenario where a large fraction of the plasma current is induced non-inductively, i.e. without making use of the transformer. This is currently a rather speculative option.

Device Operation Mode	**DEMO 1** Pulsed	**DEMO 2** Steady State
P_{th} (MW)	2200	2700
P_{net} (MW)	500	500
P_{rec} (MW)	594	600
P_{aux} (MW)	50	350
R_0 (m)	9.0	8.15
a (m)	2.25	3.0
I_p (MA)	14.1	19.8
B_t (T) on axis	6.8	5.0
f_{BS}	32%	40%
$H_{98(y,2)}$	1.2	1.3
β_N ($\beta_{N,th}$)	2.7 (2.2)	3.4 (2.8)

Table I: Main parameters of the early DEMO 1 and more advanced DEMO 2 model currently under investigation by the PPP&T Department of EFDA. Shown are the thermal output power (P_{th}), the net electrical power to the grid (P_{net}), the recirculating power (P_{rec}), the auxiliary heating power (P_{aux}), major radius (R_0) and minor radius (a) of the device, plasma current (I_p), toroidal magnetic field on axis (B_t), the bootstrap current fraction (f_{BS}), the enhancement factor H with respect to IPB98(y,2) scaling law and the normalized toroidal beta with (β_N) and without fast particle energy content, i.e. taking into account only the thermal plasma parameters ($\beta_{N,th}$)

Although ITER will bring significant advances, there remains a large gap between ITER and DEMO. Main differences between ITER and DEMO are summarized in Table II.

The power needed to drive the necessary plasma current additional to the bootstrap current for DEMO 2 (12MA) would be 480MW if one assumes a current drive efficiency of 0.05A/W and a wall plug efficiency for the heating system of 0.5 (See Section II.C). Without further improvements in alternative ways to maintain the plasma current, steady-state tokamak operation is a real challenge. A quasi-continuous tokamak operation was shown in JET and HT-7A and ISTTOK.

II. Technological Development Needs for DEMO

II.A Divertor concept

Of great importance is the design of the divertor. The power load to the divertor in DEMO can be estimated as follows. The area A of power deposition at the divertor targets can be approximated by $2\pi R\ \lambda_{SOL}\ F_{exp}$, where λ_{SOL} is the power decay length in the midplane scrape-off layer (SOL) and F_{exp} is the flux expansion from midplane to divertor targets. For DEMO with R ~ 9m, F_{exp} ~ 3-4 and λ_{SOL} ~ 0.01m this results in A ~ 2.3m^2. We took the value of λ_{SOL} ~ 0.01m as a first approximation from existing experiments, as it seems to depend only weekly on the size of the machine. The power arriving at the targets is the sum of the additional heating power and the alpha power from the fusion reactions. With an alpha power between 400 and 500 MW, and an additional heating power between 50 and 350 MW, one then finds a power flux density orthogonal to the divertor target plates between 200 and 370 MW/m^2. This number can be reduced by tilting the target plates, e.g. over an angle of 20°, leading to 190 and 350 MW/m^2 or a bit lower with further optimization, of the angle of inclination. Nevertheless, the numbers obtained in that way are still far in excess of the material limit of 10-20MW/m^2.

The first option is a conventional ITER like divertor combined with high radiation in the plasma edge to spread the heat load as homogeneously as possible over the much larger first wall surface. A possibility could be offered by seeding with appropriate impurities up to radiation fractions around 90-95%. Impurities that are currently under investigation in divertor tokamaks are N_2, Ar and Ne or a mix of them. The question is however if such regimes are compatible with sufficiently good fusion performance. Main topics for investigation are thus: (i) the effect on confinement of the seeded impurities, (ii) the effect of penetration of the impurities to the plasma centre (and/or how to avoid the pollution of the centre by the seeded impurities) and (iii) the stability of the discharge, because of the closeness to the radiation limit. It is not clear at the moment if a satisfactory solution can be found meeting all these requirements, and thus alternative solutions have to be explored as well.

ITER	DEMO
Experimental Device	Close to commercial plant
400s pulses Long interpulse time	Long pulses, high duty cycle or steady state
Many diagnostics	Minimum set of diagnostics only needed for operations
Many H&CD systems	Reduced set of H&CD systems
No T breeding required	Self sufficient T breeding
316 SS structural material	Reduced activation structural material
Modest n-fluence, low dpa Low material damage	High n-fluence, high dpa Significant material damage

Table II: Main differences between ITER and DEMO

A more advanced option consists in using innovative divertor configurations, aiming at increasing the area of power deposition.

A very early innovative concept was the doublet [7] which later was implemented in the Doublet-I (1968-1969) and subsequent Doublet-II and Doublet-III devices in General Atomics, San Diego, USA. Recently several other options have been proposed:

• X-divertor [8] where using additional coils two more X-points are created close to the targets, to further open the flux lines and spread the power over a larger target plate area;

• Super-X divertor [9], a modification of the previous concept to move the outer strike point to a larger major radius. This allows not only to increase the area of power deposition (which is mainly proportional to the major radius at the divertor target due to the near constancy of λ_{SOL} as a function of machine size) but also to increase the distance between the targets and the plasma, and thus improves impurity screening;

• Snowflake divertor (SF) [10], named after its shape around the X-point resembling a natural snowflake. It can be generated by three toroidal currents located at the corners of an isosceles triangle: the plasma current itself and two divertor coils. This concept increases not only the area of power deposition by the larger connection lengths and perpendical transport but also increases radiation because of the larger divertor volume.

• However the SF configuration is unstable: slight variations in the currents of the coils lead to another configuration with a double null; this is the quasi snowflake (QSF) divertor [11,12]. If the distance between the two nulls is small, then the properties of the QSF are close to that of the SF. The question is then what is the optimal distance between the nulls. A detailed assessment is given in [13].

• Recently also the cloverleaf divertor [14] has been proposed, named after the shape of the full magnetic configuration resembling a four petal clover-leaf. It can be generated by four toroidal currents: the plasma current itself, two divertor coils located symmetrically around the vertical axis and one on the axis. Main recent configurations are illustrated in Figs. Ia and Ib.

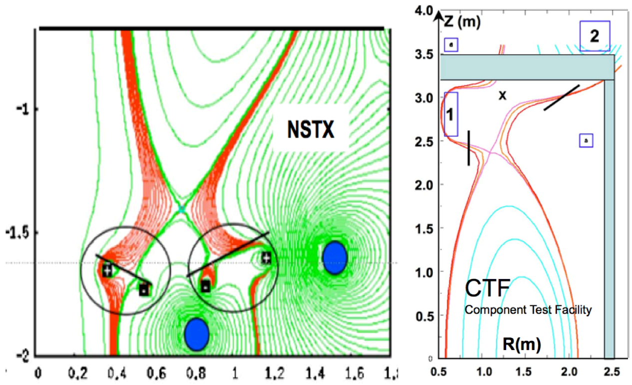

Fig. Ia: The X-divertor (left) and super-X divertor (right)

In using these alternative divertor configurations one aims at (i) decreasing the stationary and peak heat fluxes on the divertor targets and at the same time (ii) minimizing the erosion of the targets. This should be obtained by facilitating access to detachment (power and particle) by decreasing the plasma temperature below 3 eV for volume recombination to occur, improving the stability of the radiating region and increasing the wetted area [15]. At the same time (i) central plasma pollution should be avoided to minimize influence on the plasma reactivity and core radiation should be limited to allow for a sufficiently large power flux across the LCFS in order to get access to H-mode operation; (ii) the neutral particle pumping capability should be maintained. These configurations will require substantial research before becoming feasible. E.g some simulations predict that the total current in the poloidal coils could be up to ~20 times the plasma current, in case they cannot be constructed close to the plasma due to e.g. difficulties in providing sufficient neutron shielding in DEMO [16].

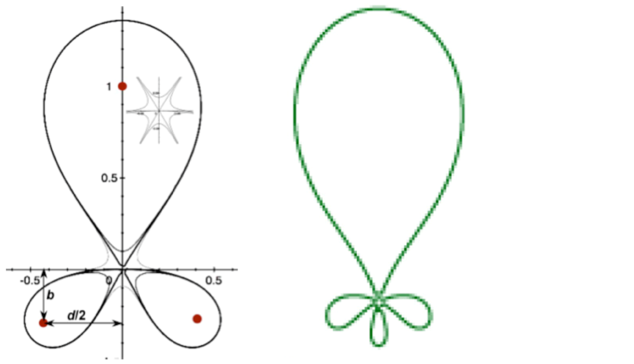

Fig. Ib: The snowflake (left) and clover-leaf (right) divertor configurations.

An additional or alternative tool to render the power exhaust capability of the divertor compatible with DEMO requirements is the use of advanced plasma facing materials such as e.g. liquid metals. The basic advantage of liquids is that they cannot be damaged by interaction with energetic plasma particles, thus showing no defect accumulation, cracking, or other surface modifications. At the same time the eroded material can be replaced in situ by e.g. capillary forces or other means. Flowing liquids offer also in principal the option to adopt larger heat fluxes using the material flow and its heat capacity. The main drawbacks are possible instabilities of the liquids in the plasma environment and the material evaporation. Candidate elements under investigation are Li, Ga, Al and Sn. Best candidates are Al, Ga and Sn because they allow for a low evaporation and sputtering rate while working at sufficiently high surface temperatures (around 1000K).

Before being able to use these two alternative options to the standard divertor in DEMO, one has to evaluate their potential "costs".

For the alternative divertor configurations this means mainly: problems arising from the increased complexity of the magnetic configuration, the compatibility with neutron shielding of the poloidal coils, constraints arising from the need for remote maintenance and increased demands on the control of the magnetic configuration.

For the liquid metals this means: avoiding core contamination with metal impurities (this implies essentially to avoid solutions that rely on evaporation), investigate potential limitations on the use of liquid metals arising from instabilities leading to splashing of the liquid in the presence of (eddy) currents and strong magnetic fields, learning how to cope with the narrow surface temperature window for operations (too high temperatures lead to too large evaporation rates).

II.B Structural materials

Defining structural and first wall materials for DEMO is another major challenge. A central issue is the material degradation due to irradiation with 14 MeV neutrons. The neutrons collide with lattice atoms, pushing them out of their equilibrium sites, leaving a vacancy and an interstitial atom. A quantity to characterize this is number of displacements per atom in the crystal lattice or dpa [17]. The migrating defects can recombine, agglomerate, form voids, interact with existing dislocations and grain boundaries etc. This leads to a number of material changes such as e.g. hardening, embrittlement, swelling and creep with the danger of losing the properties needed to guarantee the integrity of the whole device. A database providing information on the degradation of potential candidate materials thus needs to be generated.

Existing neutron sources provide only a limited answer, mainly because the average neutron energy is either too low, in fission reactors it is about 2 MeV, far below the needed 14 MeV, or too high, in the hundreds MeV range as with spallation sources. The answers can only be found by the construction of a dedicated device capable of generating the required fluxes of 14 MeV neutrons to simulate the neutronic conditions in a fusion power plant. This device should: (i) qualify the candidate materials for fusion reactors; (ii) generate the necessary data for the design, licensing and safe operation of DEMO; (iii) deepen the fundamental understanding of the radiation response of materials to high flux and energy neutron irradiation (that would allow the design of new materials for future reactors). The current proposal for such a device is IFMIF [18] (International Fusion Materials Irradiation Facility, Fig. II) in the framework of the Broader Approach Agreement between Japan and Europe.

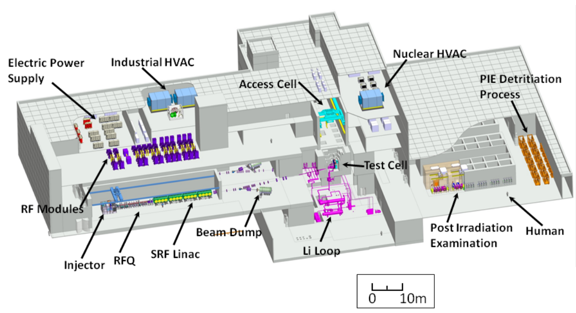

Fig. II: Overview of the IFMIF 14 MeV neutron facility for irradiation studies.

IFMIF is an accelerator-driven source of neutrons, using $^{nat}Li(d,xn)$ nuclear reactions (where ^{nat}Li represents natural Lithium consisting of 92.6% ^{7}Li and 7.4% ^{6}Li) to produce 14 MeV neutrons that simulate the conditions in flux and neutron spectrum for the first wall of DEMO and ensuing Fusion Power Plants. The symbol $^{nat}Li(d,xn)$ represents a whole set of reactions (e.g. [19, page 38]) that fall in several categories: so-called deuteron break-up reactions (where the deuteron falls apart in proton and neutron, with both half the energy of the deuteron), deuteron stripping reactions [20] (sometimes also confusingly called break-up reactions, in which one nucleon of the deuteron is stripped off, leaving a free proton or neutron at half the energy of the incoming fast deuteron), nuclear reactions in which a new (instable) nucleus is formed (e.g. $^7Li(d,n)^8Be$, $^6Li(d,n)^7Be$) and "evaporation" reactions in which neutrons from the Li nucleus are ejected (e.g. $^7Li(d,n\alpha\alpha)$, $^7Li(d,np)^7Li$, $^7Li(d,nn)^7Be$, $^7Li(d,nd)^6Li$) [21]. Each of those reactions is characterized by a different output cone for the resulting neutrons. Nuclear reactions have the broadest cone, as in the centre of mass reference frame, the reaction products have an isotropic velocity distribution. The stripping reactions result in the most narrow output cone, and are also the main source of 14 MeV neutrons in IFMIF [20, 22].

Material irradiation experiments require stable, continuous irradiation with high availability. IFMIF will achieve this using two 40 MeV, continuous wave (CW) linear deuteron accelerators, each delivering 125 mA beam current, thus resulting in two accelerated deuteron beams of 5MW. Both beams strike a concave flowing lithium target under an angle of 9º, with a footprint of 200mm x 50 mm. About 6% of the collisions result in a neutron [23], thus providing an intense neutron flux of about 10^{18} n/m^2/s with a broad energy peak at 14 MeV [24]. As the energy deposited in the target is about 1 GW/m^2, a value that cannot be supported by any solid target, it consists of a flowing liquid. The heat is evacuated with the liquid lithium, which flows at a nominal speed of 15 m/s at a temperature of 523 K. The average temperature rise in the liquid is about 50 K during its 3.3ms crossing of the two 5MW beams. The liquid is again cooled to 523 K in a quench tank using a series of heat exchangers. The inventory of liquid Li in IFMIF is about 10m^3. Many more very interesting recent technical developments can be found in reference [25].

The neutron flux in the test area falls off with distance from the lithium target, and the highest-value regions can be characterized as providing a damage production rate > 20 dpa/y in a volume of 0.5 liter capable to house around ~1000 testing specimens in 12 capsules independently cooled with He gas.

IFMIF, presently in its Engineering Validation and Engineering Design Activities (EVEDA) phase is validating the main technological challenges of the accelerator, target and test facility with the construction of full scale prototypes [26] (a deuteron accelerator at 125 mA and 9 MeV; three different lithium loops (Brasimone (ENEA), Oarai (JAEA) and Osaka University); a High Flux Test Module and He cooling gas prototype in KIT and Small Specimens Test Technique in Japanese Universities). Concurrently, an IFMIF Intermediate Engineering Design Report has been prepared to allow the construction of IFMIF on time and schedule within less than one decade whenever the Fusion community demands a fusion relevant neutron source indispensable for the next steps after ITER.

II.C Heating systems

Heating DEMO will require important further physics and technological progress on all heating systems currently in use on large tokamaks. All of them have advantages and disadvantages and at this time none of them should be excluded for DEMO since a careful assessment can only be reasonably done after ITER.

To a large extent, the size of the device dictates new needs. Taking the example of NBI, and as explained furtheron in this paper, if one would use the current systems as e.g. used at JET, with an acceleration voltage of up to ~120 keV, one would only penetrate a fraction of the minor radius of DEMO (and also ITER). Much higher particle energies are needed in the range of 1-2 MeV to deposit close to the plasma core in DEMO plasmas. But this new requirement implies that the current positive ion acceleration technique cannot be used, as the neutralization efficiency is very low at high acceleration energy. Instead negative ions have to be used and thus we are faced with the following major challenges for NBI: how to efficiently produce negative ions, how to design an accelerator in the MeV range and how to increase neutralization efficiency using a gas target as neutralizer? (the possibility of using photo-ionization is under investigation [27]). This is just one example that should illustrate the difficulties of extrapolating current knowledge to a reactor.

	GENE-RATION EFF. η_{CD}	CD Efficiency γ_{CD} (10^{20} AW^{-1} m^{-2})	WALL PLUG CD EFF. η_{CD} x γ_{CD}
ICRH	60-70%	0.23-0.32 Central or off-axis deposition	0.14-0.22
LHCD	40-50%	0.3 Off-axis deposition	0.12-0.15
ECRH	20-30%	0.35-0.40 Central deposition	0.07-0.12
NNBI	20-40%	0.3-0.45 Central deposition	0.06-0.18

Table III: Overview of the current status of auxiliary heating systems in terms of the Generation Efficiency η_{CD} (Fast Neutrals, Waves), Current Drive (CD) Efficiency γ_{CD} and the efficiency from wall plug power to plasma current η_{CD} x γ_{CD}

An important characteristic of heating systems is the potential for substituting in part the plasma current that normally is induced by the transformer, in view of extending the pulse length (in case of the pulsed DEMO 1) or steady state operation (for DEMO 2). This is expressed by the so-called current drive efficiency γ defined as $n_e R_0 I_{CD} / P_{CD}$ with R_0 the major radius (m) or the tangency radius in case of NBI, I_{CD} the magnitude of the driven current (A), P_{CD} the auxiliary heating power used (W), and n_e the plasma density (in 10^{20} m^{-3}). As DEMO 2 is rather speculative, with more special CD requirements, only the DEMO 1 case is considered below.

An overview of the present expectations for the different heating systems is given in Table III. The CD efficiency values are taken from [28]. They are computed for conditions optimized for DEMO 1. In this table the generation efficiency (second column) is taken from existing systems, notwithstanding the fact that the sources assumed in the computations of γ_{CD} do not exist for ECRH (250-280 GHz) and NBI (1.5 MeV). Of final importance is the wall plug power efficiency to generate current, and a figure of merit is shown in the last column. From the requirement of minimization of the recirculating power in a reactor, it is clear that further work is needed on all systems to improve this number.

It should also be noted that: (i) the physical mechanisms leading to off-axis current drive by NBI are not fully understood [29]; (ii) LHCD is only depositing in the edge ($\rho > 0.7$-0.8); (iii) that ECRH and NBI are more 'robust' to couple the generated power to the plasma, as power deposition is not so much depending on the edge plasma profiles as in the case of ICRH and LHCD.

For NBI, a huge effort is being put in the development of the high energy, high power (2 x 16.5MW, 1 MeV acceleration voltage) neutral beam injectors for ITER. To this end the PRIMA (Padova Research on ITER Megavolt Accelerator [30]) lab is under construction with as main experiments MITICA (Megavolt ITER injector and Concept Advancement [31]) and SPIDER (Source for Production of Ions of Deuterium Extracted from an RF plasma [32]). For DEMO continued efforts will be needed, building on the experiences from ITER, to reliably and efficiently accelerate and neutralize particles at energies between 1 and 2 MeV.

For ECRH, long pulse high frequency (250-280 GHz) sources with improved efficiency need to be developed. Existing sources (at lower frequencies ~100-140 GHz) have currently a rather low efficiency for wave generation (20-30%). This could be increased, possibly by recovering the electron beam energy in the gyrotron in "depressed collectors" [33]. A possible drawback of the use of ECRH is the large amount of stray radiation that occurs in case of badly absorbing plasma scenarios, as then several 10s of MW of microwave power will be 'sloshing' around in the device and finally arrive at first wall components, inside the diagnostic ports, etc… causing potentially large local damage (melting, burning).

In the case of ICRH the main development need is in improving the coupling of the waves over the (large and evanescent) gap between the antenna and the Last Closed Magnetic Surface of DEMO plasmas. This could imply to go to higher frequencies (~200 MHz) and/or special gas fuelling techniques in the edge to provide a propagating layer of gas in front of the antenna, without perturbing the confinement performance of the burning plasma of DEMO. LHCD has similar problems of large distance coupling. In addition, due to the large density and temperature in DEMO, the wave absorption occurs very close to the edge limiting its possibilities for driving current in the plasma core.

An important parameter to take into account in planning heating systems for future devices is the amount of power that is deposited in the plasma center. A comparison between ICRH and NBI is instructive in this discussion. A good approximation to the 1/e length of penetration of a neutral particle beam is given by

$$L_{NBI} \sim E_{NBI} / [180 \times (1+\delta(E,n_e,Z_{eff})) \times A \times n_e]$$

with E_{NBI} the energy of the injected neutral atoms (in keV), A the atomic mass of the injected atom (in amu), and n_e the line-averaged density (in 10^{20} m^{-3}). Multistep ionization is taken into account by the factor $\delta(E,n_e,Z_{eff})$ [33b]. For JET, with $E_{NBI} \sim 120$keV, $n_e \sim 5\times10^{19}$ m^{-3} and $\delta(E,n_e,Z_{eff}) <<1$ one finds $L_{JET} \sim 0.7$m, close to the value for the plasma radius $a_{JET} \sim 0.9$m. If such a system would be applied for ITER, with expected $n_e \sim 1\times10^{20}$ m^{-3}, one finds for $L_{ITER} \sim 0.35$m using the same value for $E_{NBI} = 120$keV, i.e. only about ¼ of the minor radius of ITER ($a_{ITER} \sim 2$m), In other words, mainly the outer edge of the plasma is heated. To reach the plasma centre of ITER, the injected particle energy has therefore to be increased. For DEMO, the voltage requirements are even higher to reach the centre.

	$E_{NBI, D}$	L_{NBI}	Edge NBI power deposition $P_{abs}(r/a \geq 0.5)$	Central NBI power deposition $P_{abs}(r/a \leq 0.5)$
ITER	1 MeV	1.4m	63%	37%
DEMO	1 MeV	1.4m	76%	24%
	2 MeV	2.8m	51%	49%
	3 MeV	4.2m	60%	39%

Table IV: Values for the edge and central power deposition for various values of the injected energy of neutral D atoms in ITER and DEMO. To deposit dominantly in the plasma center, very high injection energies are required.

Tabel IV gives an overview of the off-axis, centrally deposited and shine-through power for ITER and DEMO, for various values of E_{NBI} (tangential injection). Shine-through power is less than 1% for all cases mentioned.

The situation is totally different of ICRH (and also for ECRH). This heating method, on the contrary, offers the possibility to deposit a significantly larger fraction of launched power closer to the center. Indeed, the fraction of power deposited in the outer shell (r/a > 0.5) can be made less than 10%, in clear contrast to NBI as illustrated in Table IV. In fact, this can be further optimized. With a proper choice of wave frequency and heating scenario, more than 80% of the launched ICRH power can be absorbed within the plasma zone r/a < 0.2.

II.D Tritium Self Sufficiency

The Test Breeder Blanket Module (TBM) is an essential concept in the development of a future commercial reactor such as DEMO.

Any future fusion power plant reactor which will exploit the deuterium-tritium (D-T) fusion reaction to produce energy, needs to be tritium self-sufficient. Indeed, although D is relatively easy to find in sea water (its natural abundance is 0.015 %) tritium does not exist as such, and therefore, it has to be generated artificially. The easiest way to get tritium is to recover it from the Heavy Water fission Reactors (HWR) where it is produced as a by-product. Presently we estimate that about 1.7 kg per year can be produced from the Darlington Tritium Removal Facility in Canada and another 0.7 kg per year from similar South Korean reactors.

However, the operation of a commercial fusion power plant, such as DEMO, operating at the GW fusion power level, will require much more tritium. Indeed, per GW produced (thermal) power, about 55 kg tritium are needed for a full power year (FPY), or ~0.150 kg tritium per full power day.

Understandably, there is not enough tritium for a commercial fusion machine and therefore, every future fusion power plant will have to breed its own tritium needs. Therefore, one of the major objectives of ITER is to demonstrate that the current blanket technology is able not only to breed tritium, but also to extract and purify it before injecting it back into the fusion machine.

Tritium breeding requirements are quite demanding, as the process is based on the nuclear reaction between the neutron generated by the fusion reaction, taking place in the plasma and the lithium based compound, filling the blanket surrounding the torus. There are two possible ways to produce tritium in the blanket. Either by the neutron-alpha (n,α) reaction on ^{6}Li, or by the $(n, n'\alpha)$ reaction on ^{7}Li, both lithium isotopes have a natural abundance (92.4% and 7.6% respectively). To increase as much as possible the efficiency of the above mentioned nuclear reaction the blanket must contain not only lithium based ceramic material but also a neutron multiplier.

According to the current road map toward production of fusion energy, ITER might be the only opportunity for testing mockups of breeding blankets, called Test Blanket Modules, in a real fusion environment [34]. For this purpose, three equatorial ports will be dedicated to the test TBMs where each TBM port will receive two independent TBMs.

At present, the following six independent TBM Systems are foreseen for tests in ITER [35]:

- the Helium Cooled Ceramic Breeder (HCCB) and the Lithium Lead Ceramic Breeder (LLCB) for installation in Equatorial Port #02.
- the Helium Cooled Lithium Lead (HCLL) and the Helium Cooled Pebble Bed (HCPB) for installation in Equatorial Port #16;
- the Water Cooled Ceramic Breeder (WCCB) and the Dual Coolant Lithium Lead (DCLL) for installation in Equatorial Port #18;

Since the nineties the European Breeding Blanket Programme has been developing two DEMO relevant blanket concepts, the helium cooled pebble bed and the helium cooled lithium lead. For both concepts the use of lithium as breeder material is being proposed, but while the HCLL Blanket uses liquid lead as neutron multiplier, the HCPB employs beryllium. Both concepts are Helium cooled and the use of martensitic steel as structural material [36] is being considered.

In order to attain the tritium self-sufficiency, the Tritium Breeding Ratio (TBR) needs to exceed unity (best >1.1). The TBR is the ratio between the T produced in the blanket to T consumed in the plasma. The TBR value should be very accurate, as an uncertainty as small as 1% translates into 1–2 kg of T per FPY for 2–3 GW fusion power [37].

Although tritium production is an essential factor to take into account, the *Tritium extraction* operation is not less important. Indeed, the tritium bred by neutron capture in a lithium-containing blanket has to be continuously extracted by a closed loop operation and then removed from the loop for its subsequent re-introduction into the machine.

In this respect, there are several ancillary systems foreseen to carry out these operations, which are briefly described below. Firstly, the Tritium Extraction System (TES), which is foreseen to extract tritium from the lithium ceramic beds and beryllium multiplier. The TES will operate with a low-pressure helium stream (0.11MPa) and will contain approximately 0.1% pure hydrogen. The addition of hydrogen into the helium stream is absolutely necessary as it enhances the tritium release by the isotope exchange mechanism. Under such conditions the TES accomplishes the tritium extraction from the ceramic blanket in the two main chemical forms, HT and HTO. The subsequent separation, from the purging helium gas, of all diluted tritiated gaseous components, independent of their chemical form (HT, HTO, CH_3T, etc) constitutes the *Tritium removal* operation. Finally, after an ultimate chemical processing the tritium will be recovered in the Isotope Separation System (ISS), before it is sent downstream to the Storage and Delivery System (SDS), which is the main system feeding the machine with deuterium and tritium.

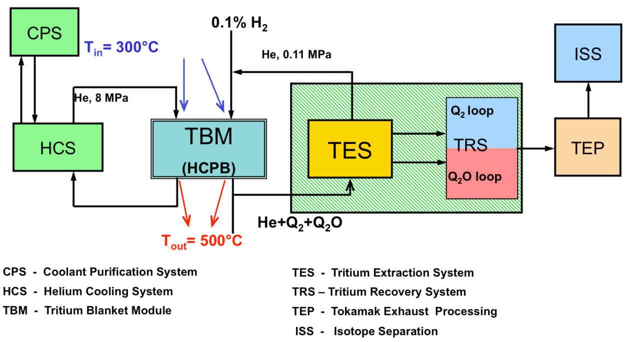

Fig. III. Flow diagram of the TES, HCS and CPS of the Test Blanket Module for ITER

Beside the TES the blanket has also to be featured with another high pressure He loop. Indeed, the thermal power, of around 3 GW (in DEMO), is extracted by means of the Helium Coolant System (HCS). In the HCS the He flows at a pressure of 8 MPa through the first wall and blanket cooling plates, which are made in EUROFER 97 martensitic steel. The inlet and outlet temperatures of the primary coolant are 300 and 500°C respectively. At such temperatures non-negligible tritium permeation cannot be avoided from the blanket modules into the He primary coolant (HCS) [37a]

Consequently, a complementary closed helium loop called Coolant Purification System (CPS) must be designed in order to remove the tritium permeated into the coolant stream. The tritium removal from He coolant has also the beneficial effect to keep the tritium inventory low in the HCS, minimising the tritium release (i) into the reactor vault in the case of a LOss of Coolant Accident (LOCA) and thus also (ii) in the secondary water-steam circuit through the steam generators. For this reason, a deep and critical analysis of the possible candidate CPS processes has to be carried out, in view of the selection of the most appropriate system and its engineering design.

A possible flow diagram of the main tritium processing systems for such a blanket concept (HCPB or HCLL) is shown in Fig. III.

After ITER important steps still have to be taken before arriving at a design for DEMO. Compared to ITER, the DEMO requirements are more demanding: the surface heat flux is about a factor of two larger, the first wall irradiation damage about 30 times larger, the neutron wall load about 3 times larger, and the local (i.e. not the full blanket) tritium production up to a factor of 10 higher [38].

II.E Diagnostics for DEMO

This paragraph raises a very important and challenging problem. Indeed many diagnostics which are currently operating in present days tokamaks and are straightforwardly being adapted to be operating in ITER cannot be directly transposed to DEMO. Worse even, many of them will simply not be working during high duty cycle burning plasma experiments [39]. Indeed the application to a steady state thermonuclear burning plasma environment induces many problems to fusion diagnostics mainly due to radiation damage, deposition of dust, influence of alpha particle bombardment, tile erosion, high heat fluxes and neutron fluence [40].

The most simple but essential diagnostic in tokamaks, the magnetics diagnostic, will already be strongly affected by radiation-induced conductivity effects in their insulators, in particular those close to the plasma that will be used for identification and control of fast instabilities. Sometimes it will be not possible to get any measurement from magnetic diagnostics depending on their specific location in the torus. An alternative solution for the measurement of the plasma shape could be to use another diagnostic that is not directly used for that purpose like tomographic Soft-X-Ray (SXR) measurements or bolometers. But this then implies a strong shielding against radiation. For a fusion reactor it is also essential to develop techniques for the detection of dust particles and erosion of the first wall (among others, also to avoid water leaks etc). These techniques are starting to be applied to laboratory samples but we need to expand these techniques in order to survey larger areas, using e.g. articulated beams. The α-particles from fusion reactions, the main source of heating in DEMO, will need to be diagnosed accurately, including their spatial distribution in the plasma. But this is a measurement that is not developed in present devices and where we have a substantial need for further developments. The measurement of escaping fast particles is also a very important problem to tackle, as the techniques currently being employed are extremely difficult to extrapolate to burning plasmas. More problematic even are all the diagnostics based on the use of fast injected neutrals from neutral beams. As stated above in Section II.C, depending on the plasma size, much higher acceleration voltages are needed, still not accessible by current technologies. Limits to the maximum acceleration voltage could thus limit the penetration of the fast particles and this would mean that the radial extent of ion temperature, rotation and current density profiles etc. would be rather limited. Also here, new measurement techniques need to be developed and tested in actual devices. More general, backscattered neutrons will affect the signal to noise ratio of practically all the diagnostics and their associated electronics and proper shielding or transfer of the information to a safe distance from the plasma using optical systems will be required [41]. But the reduced transparency of optical fibres under neutron irradiation should be strongly reduced. More general, the influence neutron irradiation on all optical elements used in various diagnostics, as well on the reflectivity of mirrors will need to be studied, maybe in the new high energy neutron generation facilities like IFMIF (see Section II.B) On top of this we have to consider the difficulties related to the acquisition of data in discharges that are lasting for days or weeks, compared to the current maximum discharge time of several tens of minutes. One important consequence of long pulse operation will be the need for a regular calibration of the diagnostics, e.g. to be triggered on demand. Indeed, the hostile environment in a fusion reactor is expected to cause severe drifts of the electronics and accelerated aging of various diagnostic components used for detection. This would imply for example calibration sources of various kinds to be permanently installed on the tokamak and remotely manipulated.

Progress in fusion energy science is strongly linked with diagnostic developments in order to measure the necessary data needed for checking the theory. Diagnostics are thus essential to further improve our understanding of the physical mechanisms and properties of the plasma. Because of the challenges sketched above, it could well be that unfortunately, we are forced to accept that only a limited number of diagnostics in DEMO will be present. This then has an immediate and important consequence for feedback control of the plasma: indeed diagnostics are the sensors that are providing real time information required for the plasma control systems to steer the different actuators (heating power, current drive systems, fuelling, plasma positioning, etc). Controlling in real time the DEMO plasma to maintain a safe and reliable plasma performance is thus becoming a challenge with a limited set of sensors. Most importantly, it is questionable whether plasma profile control, necessary for advanced tokamak scenarios [42] can be achieved in DEMO due to the rather high number of sensors needed for the real-time reconstruction with sufficient spatial resolution of the plasma equilibrium [43]. One potential solution that needs to be tested in actual devices (preferably in ITER) is to develop and validate interpretative and predictive modelling tools that could be used for the control systems. This then requires the development of synthetic diagnostics to validate the reconstruction through comparison with the limited set of diagnostic measurements on a DEMO device. In ITER such techniques should be developed, tested and optimized. In any case, even if limited, robust diagnostics will have to be implemented. The development of diagnostics providing simultaneously several informations on the plasma behaviour should be encouraged. A typical example could be the SXR diagnostic cited previously which could possibly be used to study impurity transport and MHD instabilities but – if combined with tomographic reconstruction – could provide information about the plasma shape, magnetic axis, photon temperature etc… It is clear that there is a huge need for an extensive R&D programme focused on the specific problems of implementing diagnostics in the harsh environment of a fusion reactor.

III. Physics Needs for DEMO

III.A High Density Operation and Plasma Fuelling

High density operation is an evident advantage for any fusion reactor but it is challenging. There are several density limiting mechanisms, but there is no comprehensive theory. One of the main limits is the so-called Greenwald limit [44] which is based on the empirical observation that it is difficult to run the device above a line integrated density defined by $n_e = I_p/(\pi a^2)$, with ne the line-averaged central density (in 10^{20} m^{-3}), I_p the plasma current (in MA) and a (in m) the minor radius (or minor axis of the elliptic cross-section). Moreover, in H-Mode the density profile is usually rather flat. To arrive at a peaked profile, pellet injection may provide a possible option that should be explored. For application to DEMO, optimization of the size and the speed of the pellets will be needed in order to penetrate the plasma sufficiently to fuel beyond the pedestal region of the plasma profiles.

III.B MHD Stability

Reactor requirements for high β values arise from two major considerations: high fusion power and high bootstrap current fraction. In advanced scenarios, it is generally assumed that operation is possible near the ideal magneto-hydrodynamic (MHD) limit (the so-called "no-wall limit"), usually stated as $\beta N = 4 \times l_i$ with β_N the so-called normalized beta value (i.e. the value of β normalized to the Troyon limit value for β [45]) and li the internal inductance of the plasma current. However, the real β limit could be significantly lower, due to the presence of neoclassical tearing modes (NTMs) or resistive wall modes (RWM). To mitigate this, a stabilizing wall and active feedback would be required [46, 47]. In the hybrid regime, the assumption is usually made that β_N values up to 3.5 can be sustained, thereby not exceeding the "no-wall limit". Also here neoclassical tearing modes could cause limitations, and control using localized electron cyclotron current drive (ECCD) would be needed.

III.C Alpha particle physics studies

ITER will be the first fusion experiment where a large amount of fusion alphas will be present during the high performance deuterium-tritium experiments. A simple formula relates the fraction of self-heating f by alpha particles to the Q value of the plasma : f = Q/(Q+5). With Q=10 in ITER, we find f = 66%; with Q=30-50 as expected for DEMO, f = 85-90%. In a tokamak plasma there exists a series of discrete Alfvén eigenmodes, in the frequency gaps of the Alfvén wave continuum [48]. These gaps can be due to toroidicity, elongation, triangularity, helicity etc… Frequency gaps are important because radially extended, weakly damped modes that are not subject to continuum damping can exist in these gaps, resulting in the so-called Toroidal Alfvén Eigenmodes (TAEs), Helicity-induced Eigenmodes (HAEs) etc…). Alfvén modes can become unstable if resonances occur between the velocities of the energetic particles above the Alfvén velocity $v_A = c/\sqrt{(\mu_0 \rho)}$ (where c is the speed of light, μ_0 the magnetic permeability of vacuum, and ρ is the mass density of the charged particles in the plasma) and the wave phase velocity. These Alfvén modes can lead to the loss of the energetic alphas with possible serious damage to the first wall as a consequence.

Detection of alpha particles in a burning plasma is another important topic to be developed. A clever method is not to detect directly the alpha particle but physical effects of its presence. In JET this has been demonstrated using γ rays originating from nuclear reactions between the fast alphas and intrinsic plasma impurities. In the period when JET was still equipped with the carbon inner wall (i.e. before May 2011), there were always traces of Be present in JET due to a Be evaporation technique used to condition the wall. Gamma rays from the reaction $^{9}Be(^{4}He,n\gamma)^{12}C$ were used to indirectly detect the alpha particles [49]. With the Be wall now installed in JET, this is an evident reaction to use, and the same is true for ITER. For DEMO, one has likely to choose another reaction as Be is not an appropriate wall material for a reactor device. Using a two dimensional set of γ ray cameras around the device one would be able to perform 2D tomographic reconstructions of the alpha population in the plasma. However, the γ ray detectors should be shielded against the severe neutron emission in ITER and DEMO with special neutron filters [50].

III.D Confinement and Operational scenarios

By modifying in a clever way the current profile using current drive or by freezing the current profile in a non-relaxed state by heating the plasma early in its evolution after the plasma breakdown, some confinement improvement can be obtained. A sketch of the resulting profiles of the safety factor q are shown in Fig. IV. Profiles with strongly reversed shear s = (r/q) dq/dr correspond to advanced modes, those with a flat q profile in the center to the so-called hybrid mode. Confinement enhancement factors up to 1.4-1.5 have been obtained for the hybrid mode, and higher values for the advanced modes. However, advanced modes have a tendency to be unstable, as they are characterized by peaked pressure profiles, often leading to an excess of the β limit in the plasma centre, which then leads to triggering of instabilities.

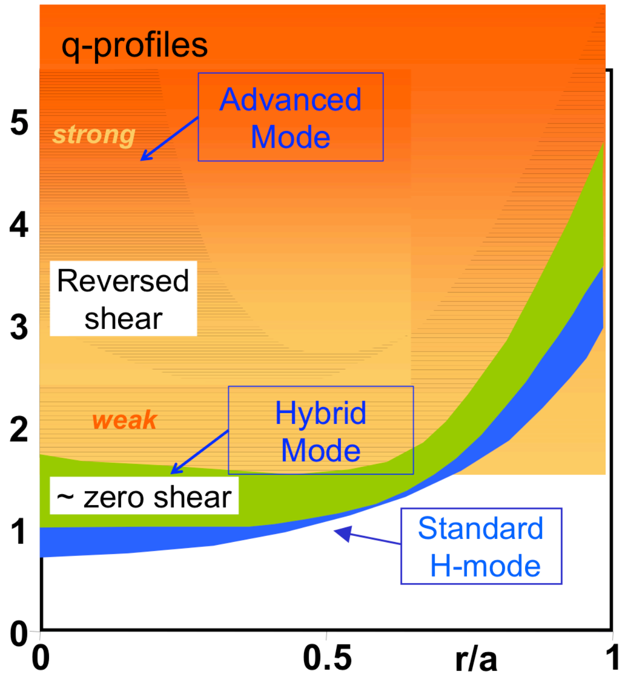

Fig IV. Profiles of the safety factor for H-Mode, hybrid-mode and advanced mode operational scenarios.

DEMO should deliver 500MW net electrical power to the grid, as discussed in the introduction of this paper. This can be done in a pulsed way or in steady-state. In many previous studies a steady-state DEMO was a primary goal. This is of course the most attractive way of operation, but it remains very challenging and requires values for e.g. β and bootstrap fraction that are hardly attainable in existing tokamaks. Advanced modes are also natural candidates for steady state operation, as the strong density gradients lead to large (non-inductive) bootstrap currents. However, they also need substantial active real-time control, which could be problematic, especially if only a limited set of diagnostics would be possible on DEMO, given that only a fraction of the power and the current is controllable in a reactor. More recent studies therefore also envisage a pulsed DEMO, with the hybrid mode as operational scenario, demonstrated by the ongoing PPP&T work in Europe (See Table 1). Advantage of pulsed operation is that the underlying operational scenarios have a much broader physics base. A major disadvantage of pulsed operation is the thermal fatigue of the first wall and structural material and the bigger size of the machine (mainly due to the larger central solenoid needed), hence a more costly device.

Both in JET and ASDEX one finds that in a majority of discharges with metallic walls confinement in H-Mode is reduced by ~20% compared to the same scenario with a Carbon wall and a similar reduction holds for the L-H threshold [51]. The reduction in confinement with respect to carbon devices can be (partly) overcome by seeding of nitrogen [52]. The reasons for these changes are not yet understood and are subject of future research. Note however that most of these discharges are obtained at a low to moderate heating power level and at rather high gas fuelling, a fact that could play a role in the observed confinement reduction. This example alone shows again that one has to be careful with straight extrapolating the current understanding to a fusion power reactor.

III.E Disruption mitigation

A disruption is defined as a sudden loss of thermal energy and particle confinement, due to a global MHD instability. It leads to a rapid decay of the plasma current, and is often preceded by a triggering MHD instability. Plasma disruptions lead to a fast and irreversible loss of thermal and magnetic energy. The energy stored in the plasma is promptly released to the surrounding structures. Large toroidal loop voltages can accelerate run-away electrons, which may hit the vessel walls, causing metallic components to melt. Elongated plasma configurations can lose vertical stability; if this occurs at full plasma current and thermal energy, it is called a vertical displacement event (VDE). When the plasma loses its equilibrium vertical position and comes in contact with the wall part of its current (known as a halo current) can flow through the wall. The average poloidal halo current contributes to the vertical force on the vessel, while the magnitude of the local halo current density puts additional requirements to the mechanical design of in-vessel components. Currents induced in conductive in-vessel components, due to the plasma displacement and/or the plasma current decay, also produce local and global forces. The deposition of energy on plasma-facing components during disruptions can have a major impact on the lifetime of these components, and it is one of the main factors that have been taken into account for the determination of the divertor plasma-facing materials in ITER [53].

The importance of avoiding disruptions in ITER and DEMO is clear: the electromechanical forces induced by disruptions scale roughly with the square of the plasma current; the runaway electron energy scales very strongly with plasma current.

The challenge for ITER and DEMO will be to limit the number of disruptions to an absolute minimum. As disruptions are unavoidable, they can be mitigated if their approach is known with sufficient warning. To this end a disruption predictor needs to be developed, which can be continuously trained, starting with a very small number of events. The most effective mitigation technique at the moment is massive gas injection [54, 55]: several bar*l of a noble gas (Ar and Ne preferred) are injected at high pressure. This is used to precipitate the quench via a radiative collapse. This results reliably in short disruptions, with consequently less opportunity to develop electromechanical loads, whereby most of the energy is lost fairly uniformly by radiation, rather than locally conducted to the wall as it would if the radiation was low both during the thermal quench and the current quench. Massive gas injection might also prevent or suppress the generation of runaway electrons, however this has not been experimentally demonstrated yet [56].

Another option is to inject a so-called large shattered deuterium-neon mixture pellet [57]. This technique consists of injecting a large cryogenic pellet (in DIII-D 15 mm diameter and 20mm long, with speeds up to 600 m/s) and shattering it into sub-millimeter fragments by impacting it on metal plates or using a curved tube. Shattering the pellet is necessary to protect the inner wall from damage by impact of a large remnant of the originally injected pellet. Shattering also helps to increase the global pellet surface area and also generates a ‘spray’ of smaller pellet particles thus increasing the ablation rate. This technique has been successfully applied to terminate plasma discharges and may be useful for suppression or dissipation of runaway electrons [58]. Before being applicable for DEMO, further work is needed. Questions to be solved are: the optimal material mixtures, size and speed of the pellets, and in how far can this can be realized with existing technology. Indeed, a fast response time could mean a close proximity to the plasma. This poses many engineering challenges to the injector technology for both Massive Gas Injection and Scattered Pellet Injection, due to the presence of strong magnetic fields and high radiation levels.

III.F ELM Mitigation

Operating ITER in the reference H-Mode scenario at 15MA and Q=10 requires good confinement, accompanied by a sufficient pedestal pressure. The strong gradients that occur in the edge region often drive strong MHD instabilities that lead to Edge Localized Modes (ELMs). The plasma energy from the pedestal region is expelled from the pedestal region in very short timescales during ELMs (~ 100μs) and can cause serious damage to the plasma facing components. In ITER it is expected that ELM energy losses could correspond to 20% of the pedestal energy, or an energy loss ΔW_{ELM} of ~ 20MJ per ELM, equivalent to a power loss in the range of 100-200 GW. In DEMO the power losses will be even larger. Mitigation of the power losses caused by these ELMs is thus mandatory. Two techniques are being pursued: (i) keeping the plasma edge conditions such that good confinement is reached, but with pedestal pressure below (but close to) the stability limits. This could possibly be achieved by applying external magnetic perturbations or could be reached in confinement regimes with small ELMs; (ii) destabilizing the plasma edge, by triggering an ELM (using external means) before the stability limit is reached. This can be done by applying a perturbation to the edge such that the ELM instability is triggered at a lower pressure than the stability limit. This technique is based on the experimental observation that the energy loss during an ELM, ΔW_{ELM}, is inversely proportional to the ELM frequency [59]. Faster ELMs lead thus to a lower energy loss per ELM.

Techniques that are currently investigated to reduce the ELM energy loss are: (i) working in a confinement regime with small ELMs [60] (ii) shallow pellet injection from the Low Field Side [61], (iii) moving the plasma up and down on a short timescale ("plasma kicks") [62] and (iv) Resonant Magnetic Perturbation (RMP) coils [63]. All of these need further research before being applicable to DEMO; questions are: (i) are confinement regimes with small ELMs transferable to DEMO? (ii) pellet injection has to be made compatible with strong neutron irradiation and handling of tritium; (iii) in how far are fast kicks a possibility, if this requires coils that are close to the plasma; (iv) the results with RMPs up to now are not very conclusive with respect to ELM mitigation, even for application to ITER; much more work will be needed there before such a system will be ready for use on DEMO; in addition it is likely that these RMP coils should sit close to the plasma to be effective, and then neutron irradiation constraints could cause difficulties to implement this method.

III.G Exhaust Pumping Systems

A fusion reactor can only successfully be operated if the ash of the fusion reaction (He) can be successfully removed. Because of the fact that the T burn-up is rather low (a few percent), the exhaust gas will consist largely in D-T fuel that has not undergone a fusion reaction plus possibly other gases used to reduce peak heat loads to the plasma facing components by edge radiation. Extrapolation of the ITER pumping systems to DEMO is not straightforward. Indeed, although they have extremely large pumping speeds, can be made tritium compatible, work under high magnetic fields and neutron loads and require relatively low maintenance (a minimum of moving parts), they have also important drawbacks for a reactor: they build up (large) hydrogen inventories with the consequent risk for explosions and thus require regular regenerations; this interferes with long pulse or continuous DEMO operations. A review of the various options indicates the need for diffusion and liquid ring pumps, together with a new type, the metal-foil pump, based on super-permeation of hydrogen and its isotopes through metallic foils, pioneered by Prof. Waelbroeck and his team (IPP, KFA-Juelich) [64]. As the metal foil pump works only for hydrogen and its isotopes it could be used to separate directly the hydrogenic fraction close to the torus, (where these could be immediately recycled) and thus could reduce to a large extent the gas throughput to the gas exhaust system [65, 66]. Such a combination of pumps is currently under investigation in the THESEUS facility in KIT, Karlsruhe [67]. An excellent overview of the current state of work can be found in [68].

CONCLUSIONS

This paper tries to illustrate the physical and technological developments that are needed before the construction of a fusion power reactor can take place. ITER will contribute to enhance our knowledge on several important aspects, but still a large step has to be taken from ITER to DEMO. The increase in size of the device a main factor driving the development needs: disruptive forces are much higher, ELM loads will be much larger, the heat load on the wall will increase, the size of the plasma is such that e.g. NBI particles need much higher energies to penetrate, etc… These are only a few examples which show that further developments are needed for currently available systems before becoming ready for use in a reactor. The need for new components, materials or systems is another factor. New physics phenomena (like the presence of a large population of fast alpha particles from fusion reactions) is a third factor contributing to continued R&D. These points clearly demonstrate that every step in fusion science is pushing the limits of what is currently known in physics and technology. In some cases even new, purpose-built laboratories have to be constructed, as illustrated in the paper.

We all know that fusion is a challenging undertaking and that patience will be needed, but it is more than worth the effort given the difficulties we are facing in the future with our current energy supply and its suspected influence on climate. It will be evidently up to you, young researchers, to tackle these interesting and very important problems. If successful, this will be your very important contribution to the benefit of all people on earth.

ACKNOWLEDGEMENTS

The following colleagues are gratefully acknowledged for stimulating and interesting discussions during the preparation of this paper: Kristel Crombé and André Messiaen (ERM-KMS, Brussels), Juan Knaster (IFMIF, Rokkasho), Gianfranco Federici (EFDA PPP&T Department, Garching), Joerg Hobirk (IPP-Garching), Francesco Gnesotto (Consortio RFX, Padova), Paula Siren (VTT, Helsinki), Volker Phillips (IEK-4/Plasmaphysik, FZ-Juelich), Domenico Frigione and Giuseppe Calabrò (ENEA, Frascati), Valeria Riccardo (JET, Culham) and Larry Baylor (ORNL, Oak Ridge).

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