The EU Strategy for solving the DEMO Exhaust Problem

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Abstract: Exhaust of power and particles is crucial for the DEMO device and the EU has developed a strategy to address the challenges. This strategy consists of a conventional approach based on extrapolation of the ITER solution (detached lower single null divertor) as well as the development of alternatives as risk mitigation. These comprise alternative magnetic divertor geometry, liquid metal targets and intrinsically ELM-free operational scenarios. On the experimental side, the EUROfusion programme has initiated both upgrades to existing linear and toroidal devices as well as plans to engage in new devices presently under construction in the EU. In parallel, the theory and modelling efforts are ramped up in a targeted effort to obtain the necessary understanding for safe extrapolation to DEMO. This is especially important for the alternatives, which cannot be tested in ITER.

1. Introduction

Exhaust of power and particles is considered a crucial challenge for the realisation of fusion power plants based on magnetic confinement [1]. For ITER, a single null quadrupole poloidal divertor with water-cooled W-targets is foreseen for this function, and the divertor plasma should be at least partially detached during high power operation. ITER plans to operate in an ELMy H-mode scenario, although Edge Localised Modes (ELMs) must be suppressed or at least strongly mitigated, which is presently foreseen to be achieved by the use of resonant magnetic perturbations (RMPs) [2]. An analysis for the European DEMO concepts clearly shows that the exhaust requirements will be different and even more stringent due to several factors:

- Larger unmitigated target heat flux q_t: in H-mode, the minimum power crossing the separatrix needs to fulfill $P_{sep} \geq f_{LH}\, P_{LH}$ (with P_{LH} the L-H power threshold and $f_{LH} \geq 1$ the operational margin). Since $P_{LH} \propto n_e\, B_t\, R^2$ [3], but the wetted area only scales like $A_w \propto R$ [4], q_t will be higher by at least $R_{DEMO}/R_{ITER} \sim 1.5$ for same B_t and n_e. Hence, the need to dissipate a large fraction of P_{sep} in the SOL to guarantee detached operation is even more stringent than for ITER. Note that same B_t and n_e at bigger R also mean operating at higher Greenwald fraction.
- Substantially higher heating power P_{heat} deposited in the plasma: aiming at a fusion power of $P_{fus} = 2$ GW and power amplification $Q \geq 30$, $P_{heat} = Pfus\ (Q/5+1)/Q$ exceeds the ITER value of 150 MW by almost a factor of 3. Consequently, the ratio P_{heat}/P_{sep} is larger by a factor of 2, and this additional power may have to be removed by impurity seeding before crossing the separatrix, calling for operation at high core radiation fraction $f_{rad,core} = P_{rad,core}/P_{heat}$ if f_{LH} is kept close to 1.
- Increased impact of neutron irradiation: due to higher fusion power and much longer operational time, the DEMO divertor components will be exposed to several dpa per full power year while the ITER divertor will accumulate < 1 dpa over the whole lifetime. This is expected to have a substantial negative impact on the thermomechanical properties of the materials used and/or their lifetime.
- Much lower tolerance to transients: while ITER plans to operate in ELMy H-mode with mitigated ELMs, due the much higher stored energy in DEMO and the much longer plasma exposure time between PFC replacement, the mitigation needs to become so stringent that DEMO practically has to avoid all ELMs [5]. Latest analysis for ITER [2] shows that ELM events should be smaller than 0.15% of the plasma stored energy, and for DEMO, this number is expected to be lower by a factor

of at least 3. This requires the development of a plasma scenario different from the ELMy H-mode with ELM mitigation.

- Increased need for first wall protection: the longer exposure time to a plasma containing high-Z seed impurities in combination with the need for thin armour on the breeding blanket makes the design of the first wall substantially more challenging than for ITER, and care has to be taken to not damage the blanket modules during ramp-up and ramp-down or in the case of off-normal events. While the ITER TBM is specified for first wall heat flux 0.3 MW/m^2, the DEMO number is about 1 MW/m^2.

For the ITER baseline scenario described above, this means that on the physics side, robust detachment control (and a method to manage an occasional re-attachment) has to be developed together with a core scenario that is compatible with a high core radiation fraction and fully suppressed ELMs. Exhaust of power and particles will also have to be managed during plasma ramp-up and ramp-down. On the technology side, it has to be ensured that the divertor components can cope with the expected heat flux in spite of the higher dpa level. The EUROfusion programme is addressing these requirements on a broad basis [6], and the progress and plans will be discussed in Section 2.

In recent years, the EUROfusion programme has started to develop alternative exhaust concepts in case the ITER solution should not extrapolate to DEMO. This programme aims at increasing the margins for exhaust and includes several elements that can probably not be validated in ITER due to technical limitations. Hence, a comprehensive R&D programme based on theory and experiment has been set up, using existing facilities and codes as well as substantial upgrades or new devices and an enhanced theory and modelling programme. In Section 3, we summarise the status of research and outline the EUROfusion development strategy for alternative divertor geometries, while the status and forward strategy for alternative divertor plasma facing units is presented in Section 4. Status and plans for the development of alternative plasma scenarios, which aim to be ELM-free ab initio, are discussed in Section 5. A summary of the strategy is given in Section 6.

2. The conventional (ITER-like) solution

In this section, we present the developments needed to adopt the ITER solution in DEMO. The elements discussed here (higher core radiation, detachment control, ELM mitigation by RMPs and advanced W-monoblock targets) are mostly testable in ITER and will hence give a large confidence for application in DEMO once this step has been taken, subject to resolving engineering and techonolgy issues for DEMO where they differ from ITER. The EUROfusion strategy for these 'conventional' solutions has been outlined in the Plasma EXhaust (PEX) strategy [7]. It aims at preparing the elements for DEMO, but rely on some level of qualification in ITER before DEMO construction can begin. Vice versa, any of the developments described below is also of great importance to prepare reliable exhaust operation on ITER. The EUROfusion assessment concluded that there is no major capability gap in the EU programme concerning the development of conventional exhaust solutions, but a continuous R&D programme is needed. This has meanwhile been implemented in the EUROfusion programme, partly also in the form of upgrades to existing devices such as WEST or the High Heat Flux facilities in Jülich.

Concerning **plasmas with a high core radiation fraction**, it is important to understand in a predictable manner the compatibility with core performance in terms of dilution and possible impact on confinement. We note that this problem cannot be treated with the usual dimensionless plasma parameter similarity approach since the radiation losses introduce an absolute temperature scale and profile effects are important. In the EU, experimentally, this is mainly studied on ASDEX Upgrade [8] and JET [9], exploring different seeding gases [10], their radiation characteristics and impact on plasma performance. On ASDEX Upgrade, $f_{rad,core}$ was even feedback controlled using bolometric measurements of the main chamber radiated power as sensor and impurity seeding as actuator [8].

Consistent with theoretical expectations, the region from where the radiation originates shift away from the cold divertor plasma with atomic charge Z, with a tendency to localize at the X-point ('X-point radiator'). It is important to note that the calculation of the total radiation in the pedestal needs to take into account the effect of charge exchange to be predictive [11]. Beneficial effects on confinement have been observed for N (ASDEX Upgrade) and Ne (JET), resulting from effects on both the H-mode pedestal and the core gradients, but this is not yet understood to a degree that would allow reliable extrapolation to DEMO. Further studies will focus on a systematic comparison of the effect of different seed impurities in devices of different size and hence plasma parameters.

Studies on the **control of detachment** also use different impurities that mainly radiate in the SOL and divertor region. While in C-wall machines like TCV, the C sputtered from the target plates under high load contributes significantly to the radiation, this is different in the metal wall devices ASDEX Upgrade and JET. Recent studies focused on feedback control of detachment, and on TCV, it was demonstrated that the position of the detachment front, using CIII emission images as a proxy, could be controlled using D gas puff as an actuator [12]. On ASDEX Upgrade, stable feedback control of the position of radiation maximum in the vicinity of the X-point was demonstrated using bolometry as sensor and impurity puff as actuator [13]. Examples for the feedback control are shown in Fig. 1. The location of this 'X-point radiator' was also found to affect the core plasma characteristics, leading e.g. to a disappearance of ELMs on ASDEX Upgrade [13] and JET [9]. These experiments show clearly that the control of core and SOL/divertor radiation cannot be fully separated and has to be treated as a whole in future. However, we note that the power levels in present day devices are not sufficient to test this combination under DEMO relevant loads since at $P_{sep} = P_{LH} f_{LH}$, i.e. high $f_{rad,core}$, the power flux into the SOL is at relatively low level compared to DEMO. Experimentally, this will only be possible to validate on ITER with an upgraded additional heating due to the relatively low Q compared to DEMO. Another question that will be addressed in the EU exhaust programme is the compatibility with good divertor compression and pumping, since the motion of the detachment front out of the divertor towards the X-point is suspected to decrease divertor closure. The experimental studies are complemented by theoretical investigations aiming at predictive capability. More and more, models are also used in 'flight simulator' type simulation of discharge control [14] including core radiation and detachment. ITER will provide an important platform to validate these schemes, albeit only in the high power phase in the 2030s.

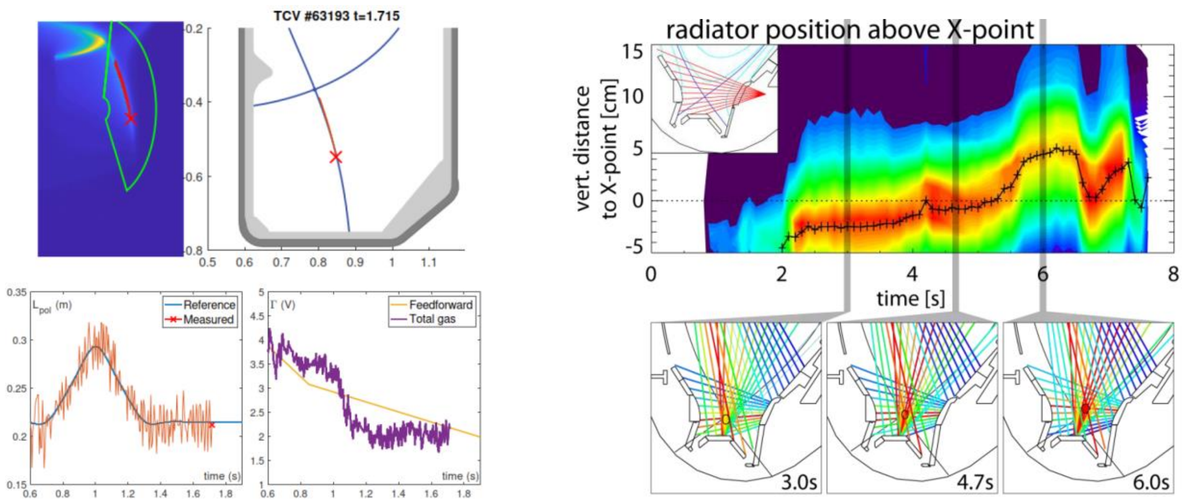

Fig. 1: Feedback control of the position of the detachment front on TCV (left) and the X-point radiator in ASDEX Upgrade (right). TCV used a reconstruction of camera images, ASDEX Upgrade bolometer lines of sight to locate the object to be controlled, gas puff (D on TCV, N on ASDEX Upgrade) was used as actuator.

The ITER strategy for **ELM mitigation or suppression** is based on the use of RMPs and pellets. Since pellets cannot be used for full suppression, and their application for ELM pacing may significantly increase the gas throughput, the EU DEMO programme has a focus on RMP suppression, or preferably avoidance (see below). In the experimental programme, both ASDEX Upgrade [15] and MAST [16] have contributed to the studies on ELM suppression, together with collaborative experiments on DIII-D in the US [17]. While the operational range for full ELM suppression has continuously widened, the understanding of the access conditions is not yet fully clear. In particular, the experimentally observed upper density limit for full ELM suppression is a concern since ITER and DEMO will operate at pedestal top densities higher than the value found in present day experiments [15]. A better understanding of the underlying physics is needed in order to predict the applicability of RMP ELM suppression in future devices, with the experimental verification coming in ITER experiments. An important contribution is also expected from JT-60SA that will access a combination of plasma parameters that cannot be studied in the present RMP-equipped EU devices. On the technology side, application of RMPs in DEMO would most likely require the use of in-vessel coil, and the EUROfusion programme also addresses the implications of this for the DEMO design (more broadly also in view of the use of in-vessel coils for position and shape control).

The **solid divertor targets** for ITER will consist of water-cooled solid W monoblocks that are designed to withstand a peak stationary power flux of 10 MW/m^2. The EU DEMO programme has started an effort to design, manufacture and test improved variants of this target that should withstand up to 20 MW/m^2 before irradiation in view of the expected stronger degradation of materials due to the much higher n-fluence in DEMO. The EUROfusion programme has started an in-depth study of the issue with the aim to quantify the impact on design rules. Recently, several variants have been successfully tested, surviving at least 500 pulses at 20 MW/m^2 with a pulse length of ~ 10 s each (sufficient to reach thermal equilibrium in each cycle) [18]. A variant similar to the ITER design was chosen as reference due to its design simplicity, but a risk mitigation option is kept in the form of a fibre reinforced design variant that promises better mechanical properties. Also, studies on He cooling as an alternative to the presently favoured water cooling were conducted. We note that the water cooled concepts could in principle be tested in ITER, e.g. in a second generation divertor, which would serve as an important validation step.

The ITER **first wall armour** solution is different from that needed for DEMO where the armour (W in DEMO versus Be in ITER) has to be much thinner with a different engineering structure, so this will need to be developed by a combination of modelling of the plasma source of radiation and fast neutrals (erosion) combined with laboratory tests (heat load and thermohydraulic). Some integration aspects can be explored with the ITER Test Blanket Modules, since even if these will be recessed from the first wall, they will see some radiative and fast ion heat flux in ITER. A set of limiters has been designed to protect the first wall elements in plasma ramp-up and ramp-down phases, where loading of few 10s of MW/m^2 can occur for a few 10s of seconds. In case of off-normal events, loads may reach up to several 100s of MW/m2 for some 10s of milliseconds, and here, protection limiters will be sacrificial, i.e. melting will occur, but without destroying the cooling channels. This concept ensures that the heat load on the regular (blanket) first wall elements will always stay in the range 1-2 MW/m^2 [19].

3. Alternative divertor geometries

While ITER will employ a single null quadrupole X-point divertor (SND), several other alternative concepts have been proposed in the recent years. These aim to

- increase the dissipated power by increasing the volume of the radiative zone in the SOL and divertor. In the Snowflake divertor (SFD) configuration, this is achieved by increasing the region of low poloidal field around the null point applying a hexapole rather than a quadrupole poloidal field (i.e. adding two quadrupole nulls in one location). In the Super-X divertor (SXD) configuration, additional PF coils guide the divertor fan to a target at larger major radius, keeping the radiative zone away from the core. A tightly baffled long divertor leg should also increase neutral pressure in the divertor and hence ease dissipation through increased recycling and CX.

- broaden the wetted area by increased cross-field transport through increased turbulence (expected in the Snowflake configuration due to the low poloidal field in the null point region), longer connection length (expected in both SFD and SXD as well as the X-divertor (XD), where an X-point is created just below the target) and increased major radius of the target for the SXD. For the SXD and the XD configuration, it has been discussed to increase the wetted area by poloidal flux expansion. This is limited by too small a grazing field line angle that can cause overheating of leading edges in case of non-perfect alignment due to manufacturing tolerances, at least under attached conditions. In principle, a perfect SFD promises to allow for 4 strike points, but in practice the two null points will be separated and it will be difficult to control the poloidal flux distance such that it is below a power fall-off length.
- enhance positional stability of the detachment front due to flaring of the flux surfaces in front of the target in the case of XD and SXD, where the expanding fan should stabilise the motion of the detachment front up the temperature gradient that is typical for the SND in the case of the X-point radiator as described in the previous section. This should lead to a reduced impact on the core plasma performance. Theory predicts that total flux expansion is more effective than poloidal expansion [20], which would favour the SXD over the XD. Also, for poloidal flux expansion, the caveat about the grazing field line angle mentioned above holds.

All these measures aim at establishing and/or increasing the parameter window for detached divertor operation, easing the requirements imposed on the core scenario (density, impurity content) and leading to more robust controllability.

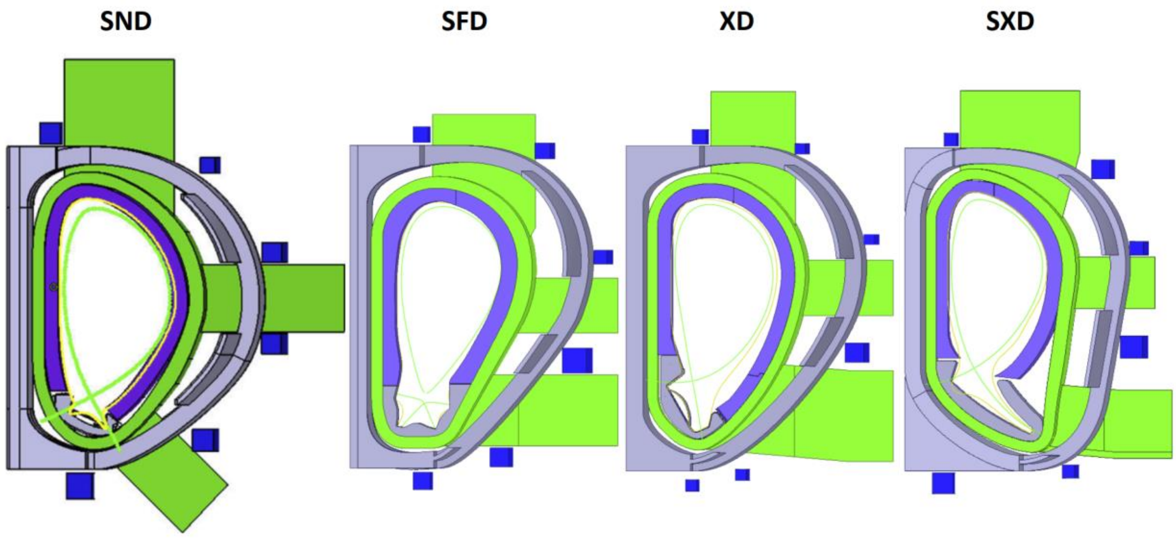

Fig. 2: Drawings of alternative divertor configurations studied for EU DEMO (Single Null (SN) reference, Snowflake divertor (SFD), X-divertor (XD) and Super-X divertor (SXD)).

Since these alternative divertor geometries usually require a special set of PF coils for their creation, they cannot be realised in ITER, and a strategy for their qualification needs to proceed on a different route than that described in Section 2, where ITER serves as an experimental verification of the design assumptions for DEMO. In its PEX strategy [7], the EUROfusion programme has hence outlined an approach to qualify alternative divertor configurations for DEMO. In a first ‘proof-of principle’ step, this approach relies on the development of adequate theoretical models to study the expected benefits of alternative divertors, an engineering study of the implications for the DEMO design, and an enhancement programme of the EU medium size tokamaks to enable the experimental study of the exhaust properties.

In the theoretical and engineering analysis, a first comparison of the SND reference with double null divertor (DND), SFD, XD and SXD was performed ([21], [22], see Fig. 2) with the following conclusions:

- All alternative divertor configurations studied can be realised using external PF coils for the present EU DEMO core plasma, but with impact on the technology (larger TF coil volume, especially for SXD, less OH flux swing at same aspect ratio, more complex remote maintenance, especially for SFD, and TF stresses sometimes exceeding critical levels, indicating the need for redesign). Vice versa, technology sets some boundary conditions for the alternative geometries (e.g. limiting the length of the outer SXD divertor leg).
- Using the present plasma models, it was only possible to study the decrease in required impurity puff to lower the target temperature to 5 eV, which was used as a crude proxy for the onset of detachment. So far, the main emphasis has been on relative differences between the configurations. These were found to be appreciable for (S)XD and the SND reference, widening the operational space. For the SFD, these studies could not yet be completed due to the complex geometry for modelling. On a down-scaled SFD simulation, it was found that an increase in turbulent transport led to an activation of all 4 strike points, but this is yet to be confirmed for the DEMO parameters.
- Configuration control has been identified as a possible showstopper: an unforeseen drop of $\Delta\beta_{pol} = -0.1$ was found to move the strike point by 10s of cms across the target in both SFD and SXD. Both configurations will also experience a comparable vertical jump of the whole plasma, and it is not clear if, in a single null configuration, this can be reduced to an acceptable level. Control power requirements will in general be large and may be prohibitive for some configurations.

These studies will be continued, using refined models that can capture important effects that were not addressable at present, including the impact on the core plasma performance, and addressing the potential benefits of the SFD concerning its operational window in a manner similar to XD and SXD. We note here that a simultaneous optimisation of physics and technology aspects may finally lead to a 'hybrid' solution combining elements from several of the above concepts [22].

On the experimental side, the main EU device for studies of alternative divertor configurations so far has been the TCV tokamak, due to its unique shaping capabilities. First studies showed that a trapping of the radiation zone in between the two X-points of a 'snowflake minus' configuration (where the second quadrupole X-point is in the private flux region) can occur, but with little impact on the total radiated power before the plasma disrupts [23]. For the XD, a 'trapping' of the detachment zone in front of the target was shown, evidenced by a higher achievable density before the detachment front moves from the target to the X-point [24]. While these results are encouraging, they suffer from relatively low power levels and the open divertor configuration of TCV, which negatively affects detachment. EUROfusion has hence started an enhancement programme for the medium size tokamak devices to enable further experimental studies of alternative divertor configurations:

- On ASDEX Upgrade, two in-vessel coils will be installed in the upper divertor, together with a flat high heat flux target and a cryo-pump, allowing to test several configurations with high heating power and very good divertor diagnostics.
- On MAST Upgrade, a tightly baffled double-null SXD has been installed that, together with extensions of heating power and diagnostics, will allow the study of this alternative in a configuration optimised specifically for this type of divertor geometry.

- On TCV, additional baffles have been installed to increase the divertor neutral pressure for a variety of configurations. This effect has already been shown in first comparative experiments [25], [26]. Furthermore, additional heating power will be installed.

Combining experiments conducted using these upgrades with the modelling developments described above, the EUROfusion PEX strategy foresees to conclude the proof-of-principle phase by making a decision on the best alternative divertor configuration to be developed further as an alternative to the SND for DEMO. At this decision point, it is foreseen to engage in the DTT tokamak [27], presently under construction in Frascati by ENEA, to use it as a testbed of the 'EUROfusion divertor'. The DTT tokamak will provide plasma parameters that are not accessible in present day experiments, even with the PEX upgrades, and should serve, together with further developed theoretical models, as the platform for validation of the alternative divertor concept for DEMO.

4. Alternative plasma facing units (PFUs)

As described in Section 2, the EUROfusion programme is developing divertor PFUs for DEMO based on the ITER solution, i.e. solid W monoblock targets, but aiming at higher heat flux handling and with emphasis on mitigating the consequences of the higher n-fluence in DEMO. While these concepts could in principle be qualified in ITER, at least as long as they are water cooled, this will not be the case for alternative concepts that rely on the use of liquid metals as plasma facing material. These concepts promise the absence of mechanical failure of the surface in the form of cracks or fatigue failure (also relieving the mechanical problems arising from n-fluence), higher tolerable heat flux due to self-replenishing of the surface (allowing thinner surface material thickness) and higher resilience to transients due to vapour shielding under strong overheating (strong evaporation leads to high radiative cooling in front of the target). Also, depending on the design, liquid metal divertor modules could provide a reduction of the very tight alignment tolerances that monoblocks will need.

In parallel to the strategy to develop an alternative divertor geometry for DEMO discussed in the previous section, the EUROfusion PEX strategy [7] hence also proposed a development strategy for alternative PFUs in the form of liquid metal targets. An analysis of the proof-of-principle phase highlighted two main issues:

- Temperature window for operation: the usable temperature window is given by the melting point (lower limit) and the temperature at which evaporation would lead to an inacceptable concentration of the target material in the plasma (upper limit). Depending on the concept envisaged (see below) this temperature window may be quite narrow compared to that of solid material.
- Tritium retention: the retention of T in the liquid metal must be such that the inventory limit of DEMO is not exceeded, and the tritium should be extractable. In general, the pumping of hydrogenic species should also not change recycling by an amount that would prohibit establishing a detached divertor solution, which is expected to be required also for liquid metal solutions.

In response to these points, the EUROfusion programme set up an effort to select a concept that promises to be compatible with these issues. Previous experiments had mainly focused on the use of Li, but in a concept not relying on evaporation, the useable surface temperature window is estimated to be quite limited, of the order of [180°C - 420°C], with the upper limit estimated from evaporation of pure Li being equal to the expected tolerable erosion flux, and neglecting any prompt redeposition (see below). There are concepts that rely on strong evaporation of Li [28], based on the experimental observation that Li does not strongly affect plasma performance, but a concept for a closed loop cycle (e.g. condensing and reprocessing) is still far from demonstrated. Such a concept would also be needed for removing the T stored in the Li, since Li has a very high T retention, and work has begun to develop this [29]. Finally, the strong retention also questions the access to a detached divertor solution, since this usually relies on strong recycling in order to avoid excessive gas flux through the machine, which would lead to a large inventory. This weighs strongly against the beneficial effects on plasma performance that are observed with the use of Li due to the lowered recycling in present day machines.

The EUROfusion programme hence studied in detail the use of Sn as an alternative to Li [30], with quite encouraging results. Due to its higher Z, the tolerable plasma concentration for Sn is about two orders of magnitude lower than for Li, and hence the erosion limit could be lower by about this amount. Still, Sn has a larger operational expected window than Li, namely [230°C - 880°C] determined by the same criteria stated above. In experiments in the Magnum PSI linear device, a strong redeposition of the eroded and evaporated material was found for Sn, which would raise the upper temperature considerably, up to as high as 1250°C. A similar effect is expected for Li and would also significantly raise its upper temperature limit (in [30], it is estimated that it could go up to almost 700°C). It must be noted, though, that measurements of D-retention in Sn found that D can be trapped in bubbles below the liquid surface [31], enhancing the D retention, albeit still much smaller than in Li. Also, micro droplet ejection from liquid Sn surfaces due to bursting bubbles has been observed in these studies, leading to an enhanced erosion rate. In these cases, the erosion as well as the D-retention of the liquid metal PFU can be appreciably higher than that of pure Sn [32], but studies are under way how to mitigate this effect (see below).

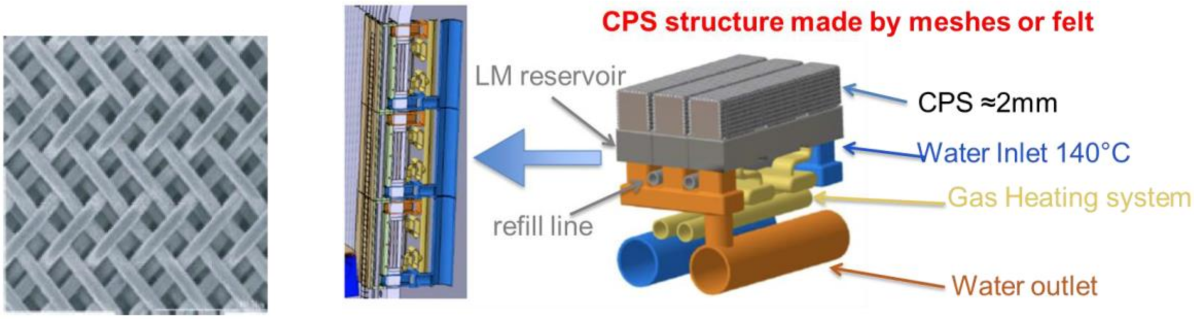

Fig 3: W mesh (left, [33]) used to contain liquid Sn in a Capillary Porous System (CPS), and CAD drawing of a conceptual design of a plasma facing unit based on this technology.

It is hence clear that ultimately, the performance of a liquid metal PFU will depend not only on the properties of the chosen material, but also strongly on the component design. Several concepts exist, involving free flow across the surface of the PFU or even, as mentioned above, evaporation and condensation in a different area of the vacuum vessel, which both promise very high acceptable thermal loads due to a large effective thermal capacity of the flowing material. However, these solutions present large technical challenges concerning the stability of the surface and hence, the EUROfusion programme has selected the Capillary Porous System (CPS) option. Here, a W mesh with pore size of ~ 50 μm is wetted by the liquid metal, creating a liquid surface tightly bound to the mesh by surface tension, with the eroded and evaporated material replenished by the capillary force. It is believed that by choosing an appropriate mesh structure and size, the problems of bubble formation and droplet ejection can be minimized [32], and this will be a topic of future R&D. Studies of the power handling capability of CPS components indicate that it should at least be comparable to that of the advanced solid targets described in Section 2, i.e. up to 20 MW/m^2, but promising more benign reaction to excursions due to the above described effects of vapor shielding and self-replenishing. A picture of a W-mesh together with a conceptual design of a Sn CPS PFU is shown in Fig. 3, still subject to optimization towards DEMO application.

Based on these promising results, the EUROfusion programme decided to go ahead and aims at building a prototype of an optimized high heat flux Sn CPS PFU, that can be tested first in linear devices, and, after confidence in the power handling capability has been gained, installed in a tokamak. While an early test can be performed by replacing a set of divertor tiles on ASDEX Upgrade using the divertor manipulator which allows this as a temporary test installation, the EUROfusion strategy envisages that at a later state, a dedicated tokamak experiment should be equipped with a full liquid metal divertor. This is presently foreseen for COMPASS Upgrade [34], which, as a high field device, will allow large heat fluxes in combination with a hot (300°C) wall for testing a full liquid metal divertor from ~2025 on, and later on possibly also in DTT. These experiments will also allow to study the erosion, migration and redeposition in a real tokamak environment. In parallel, the design will be optimized towards its performance in the DEMO environment, which requires to consider also the expected effects of combined heat and radiation loads.

5. Alternative plasma scenarios

It was discussed in Section 2 that ITER plans to suppress or mitigate ELMs using RMPs and pellets, and the prospects for adopting this strategy on DEMO were discussed. However, in view of the requirement to avoid any large ELM in DEMO, an analysis of intrinsically ELM-free operational scenarios that still benefit from an edge pedestal has been carried out by EUROfusion [35]. All of these modes involve pedestal transport mechanisms in addition to the inter ELM transport observed in type I ELMy H-mode that keep the pedestal conditions below the ELM stability limit and enhance particle transport to a level that the impurity level is tolerable (this is not the case in 'normal' ELM-free conditions which are terminated by impurity accumulation). The regimes may be differentiated by the nature of this additional transport.

In Quiescent H-mode (**QH-mode**), there is a macroscopic 3-D deformation of the plasma edge, usually rotating in the laboratory frame. This mode is called the Edge Harmonic Oscillation (EHO) and is thought to be a saturated kink mode. Experimentally, the occurrence of this mode coincides with conditions of strongly sheared edge rotation, established e.g. by using ctr-injected NBI, but there is not yet a quantitative theoretical understanding of the access conditions. On the theory side, this is a goal of the EUROfusion programme. These studies should also address the experimental observation that QH-mode has been preferentially accessed at low absolute density, but not in regimes where strong fuelling is applied to raise the separatrix density as is needed to access divertor detachment. Experimentally, it remains to be shown that QH-mode can also be established with a W-wall. While in DIII-D, equipped with a C-wall, QH-mode has a relatively wide operational window [36], in ASDEX Upgrade, which could access QH-mode with the C-wall, it has so far not been possible to establish QH-mode with the W-wall. While the EHO could be generated establishing strong edge shear, the discharges do not become stationary due to a too large influx of W, leading to a radiative collapse. It remains to be resolved if this is a general result (EHO not strong enough to expel heavy impurities), or conditions can be established to operate QH-mode in a W-wall environment. For this question, it will be important to equip the JT-60SA device with a W-wall in future, which is part of the EUROfusion strategy within the Broader Approach agreement with Japan. Similar to the present role of JET in the EUROfusion programme, JT-60SA can also be used to widen the edge parameter space towards higher separatrix density at still low collisionality, which will be the case in ITER and DEMO, but cannot be accessed in present day medium size devices.

In the **I-mode** regime, the pedestal is mainly due to a steep edge temperature gradient, while the density gradient in this region is substantially lower than in H-mode [37]. This is very beneficial in terms of keeping the edge pressure below the ELM stability limit, but may lead to a lower overall energy confinement than in H-mode. The I-mode edge is characterised by a 'weakly coherent mode' (WCM) that may be the cause of the absence of a particle transport barrier, but the theoretical understanding of the WCM is not yet mature enough to predict its access conditions or the level of transport induced by it. Experimentally, access to I-mode is favored by conditions in which the L-H transition power threshold is high, such as ion grad B drift pointing away from the divertor or high magnetic field, to assist avoidance of the density pedestal associated with H-mode. On Alcator C-mod, it was found that the heating power to access I-mode increases with density, like the H-mode threshold, but the increase with B-field is weaker, such that an I-mode operational window between L-mode and H-mode occurs at higher magnetic field. This would favour I-mode access in ITER and DEMO, but needs a theoretical foundation since it is unlikely that the absolute value of B_t will play a role. Similar to the QH-mode, it is difficult to establish I-mode operation with a detached divertor in present day devices, and experiments are targeting both experimental studies as well as theoretical development to understand the cause of this behaviour.

The EUROfusion study on ELM free regimes mainly focused on QH-mode and I-mode and identified the issues mentioned above (access conditions, compatibility with radiative mantle and detachment) as possible showstoppers, together with the quality of energy confinement. While the operational regimes are presently characterised by their confinement time relative to type I ELMy H-mode (so called H-factor), it is uncertain if their scaling with plasma parameters will follow the H-mode scaling and hence, confinement will have to be understood by physics based models allowing extrapolation with high confidence. The study also concluded that the present portfolio of devices in the EU and its international partners should be sufficient to perform the experimental studies needed. An important part of the strategy will be to prepare the validation of the operational regimes in ITER, which should in principle be possible, albeit the ITER baseline scenario is RMP suppressed ELMs (see section 2).

In addition to the two regimes discussed above, a number of other ELM free scenarios relying on an edge pedestal are presently under investigation in the EU and worldwide. One of them relies on the observation that with increased fuelling and edge radiation, as was already mentioned in Section 2, ELMs become small and very frequent until they are finally no longer distinguishable as separate events and lead to a quasi-continuous exhaust of energy and particles. It is presently conjectured that the boundary condition for this regime is the stability close to the separatrix [38], where ITER and DEMO may have similar conditions than present day experiments (as opposed to the very different pedestal top parameters). This would allow this regime to be applicable in ITER and DEMO as well. Another regime of interest is the Enhanced D-Alpha (EDA) mode that again relies on enhanced fluctuations observed in the pedestal, this time the so-called Quasi-Coherent Mode (QCM) [39]. This regime, originally discovered in Alcator C-Mod, has regained attention due to a recent systematic extension of its parameter space on ASDEX Upgrade [40]. In the EU strategy, these modes are studied on the EU devices ASDEX Upgrade and JET, with accompanying theoretical studies that aim to understand the pedestal fluctuations responsible for the stationarity of the ELM free phase. Again, these modes could in principle also be applied in ITER, allowing a stringent experimental test for their extrapolation to DEMO.

Finally, the L-mode regime that does not feature an edge pedestal and hence is intrinsically ELM-free had previously been discarded from the candidate scenarios due to its lower confinement, leading to a very large size DEMO. However, interest is now on scenarios with a plasma cross-section substantially different from the usual operating range, namely that with **negative triangularity** (NT). It was observed in the TCV tokamak that this configuration can exhibit steeper temperature gradients over the outer half of the plasma radius than observed in L-modes with positive triangularity, thus making up for the absence of a pedestal and allowing for confinement quality comparable to the H-mode [41]. While the early TCV experiments were carried out in plasmas dominated by electron confinement, more recent experiments on TCV and DIII-D point to the existence of such a beneficial effect also for the ion channel, which would make NT plasmas very attractive for DEMO [42]. In addition, no lower limit for the power across the separatrix to access the regime has been found so far, indicating that the exhaust problem could also be relaxed from this point of view. A EUROfusion assessment of the NT option [43] confirmed the possible attractiveness of the scenario, but pointed out that the parameter space in which the regime has been obtained so far is very limited, so that a concentrated experimental and theoretical effort is needed to address its viability for DEMO. It must also be mentioned that the coil and vessel configuration of an NT tokamak is quite different from positive triangularity [44], so the programme will address the technological implications in parallel. This also means that the scenario cannot be validated in ITER. Similar to the PEX strategy, there must hence be a proof-of-principle phase which can be carried out in existing devices (TCV, due to its shaping capability, and ASDEX Upgrade and DIII-D with more restricted shapes, see Fig. 4). Following this, a dedicated device (new device or major rebuild, e.g. of the Italian DTT device) would be needed for a consecutive validation phase.

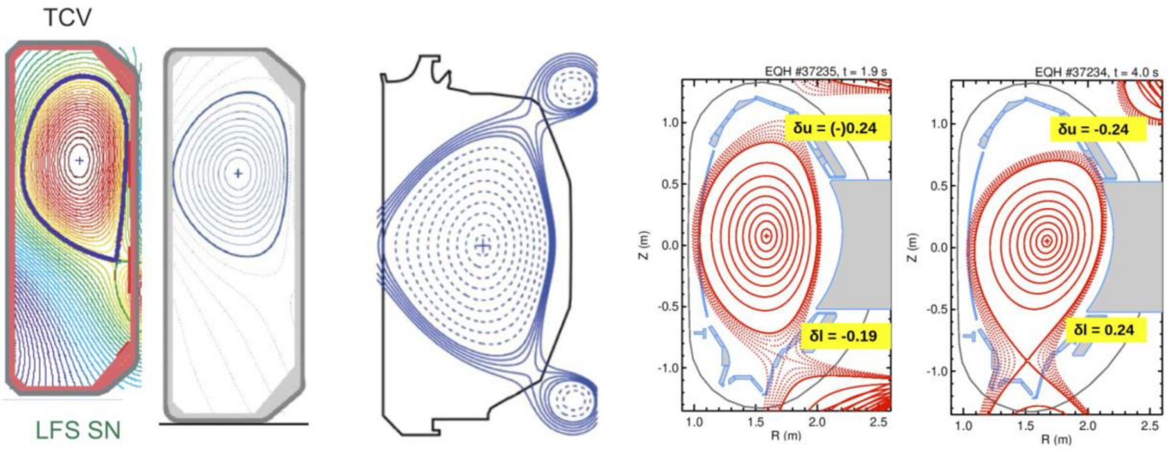

Fig. 4: Negative Triangularity (NT) equilibria realised on TCV, DIII-D and ASDEX Upgrade. In DIII-D and ASDEX Upgrade, strong NT is only possible for inner wall limited configurations, while TCV can produce a divertor configuration ('LFS SN') with strong NT shaping.

Similar to the alternative exhaust configurations discussed above, the optimal scenario for DEMO may be a combination of several of the elements described here, and independent theoretical exploration of how to achieve the desired edge characteristics sustainability is an important part of the strategy.

6. Summary and Conclusions

During the last decade, it has become clear that exhaust is one of the major challenges for the design of a DEMO device, and the EUROfusion programme has developed a broad approach to address this challenge. It consists of a baseline strategy that builds on the solution foreseen for ITER (detached single null divertor with W-monoblock plasma facing units), extending it where needed to comply with the even more challenging environment of EU-DEMO. Concerning plasma physics, promising results have been achieved concerning operation with high core radiation fraction, while in the technology area, divertor plasma facing units have been developed further, taking into account the expected effects due to the much higher neutron fluence. A concept for protecting the first wall has been outlined that promises to reliably protect the breeding blanket modules. The ITER baseline strategy will be further developed using the existing EU tokamak devices, together with JT-60SA, in preparation of the demonstration of the viability of the concept in ITER in the 2030s. ITER will provide a unique environment to test the approach in an integrated manner, combining a reactor-grade core plasma with the exhaust physics and technologies developed in the coming years.

As a risk mitigation strategy, the EUROfusion programme has started to develop alternative exhaust solutions that promise to provide ore margin, but can not be tested in ITER for technical reasons. These are alternative divertor configurations, which rely on elements such as increased connection length, additional active X-points and flux expansion in front oft he target in order to increase the operational window for detachment. For plasma facing units, alternatives using liquid metal targets are studied, and the development is aiming at qualifying capillary porous structure wetted with Sn for application in DEMO. Concerning plasma operational scenarios, intrinsically ELM-free solutions are studied. In scenarios which rely on an edge pedestal, these seek to provide enough additional transport that the plasma is ELM stable and impurities are flushed. In scenarios without a pedestal, such as negative triangularity L-modes, transport must be reduced over a larger part of the plasma to reach the required core parameters.

In the past years, the EUROfusion programme has been adapted to be ready to address these issues by combined experiments, theory and modelling. To this end, experimental devices, both toroidal confinement and linear test facilities, have been upgraded, and a new structure for addressing the theoretical challenges in a co-ordinated way has been set up. Also, EUROfusion will engage in the exhaust programme of new devices presently under construction in the EU. These will provide the basis for an exciting programme to be carried out by the EUROfusion consortium partners in the next European Framework Programme, FP9 (2021-2027).

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