First IAEA Technical Meeting on Divertor Concepts

29th September – 2nd October 2015

IAEA Headquarters, Vienna International Center
Vienna, Austria

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Meeting Website: http://www-naweb.iaea.org/napc/physics/meetings/TM49934.html
Topics

1. Divertor and Confinement

2. Radiative Power Exhaust

3. Scrape-off Layer and Divertor Physics

4. Steady State Operation and Transient Heat Loads

5. Plasma Facing Component Materials and Heat Exhaust for Steady State Operation

6. Divertors for DEMO and Reactors
**Schedule**

*Reviews (R) are allotted 35 min + 5 min for discussion*

*Invited Orals (I) are allotted 20 min + 10 min for discussion.*

*Regular Orals (O) are allotted 20 min + 5 min for discussion*

*Poster (P) are allotted 5 min + 2 min for discussion*

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Building on about 20 years of physics simulation, engineering design work and component testing, the ITER divertor, the largest and most complex tokamak divertor ever to be constructed, is now approaching the full procurement phase. A first complete design review was already conducted in 2009, at that time featuring carbon fibre composite (CFC) monoblocks in high heat flux (HHF) areas. This divertor was to be used for the first “non-active” plasma campaigns in H and He, then exchanged for a full tungsten (W) variant before nuclear operation began. In 2013 the ITER Organization (IO) decided to eliminate CFC and design an all-W armoured divertor which would be installed from the start of operations and would be expected to survive until at least the end of the first major DT campaign [1].

Although the decision to switch to full-W was based on the solid foundation of many years of physics and engineering design for the CFC variant, all metal-PFCs present different challenges. In particular, the absence of natural C radiation requires the use of extrinsic radiators to ensure technologically feasible target power fluxes and surface melting, due both to steady state and transient heat loads, has the greatest potential to compromise machine operation. New “carbon-free” simulations are thus required to investigate divertor performance and compatibility with burning plasmas, accounting for the recent, rather pessimistic experimental scaling [2], which implies very narrow (~1 mm) SOL heat flux widths for parallel power flow on ITER. An assessment of the likelihood and consequence of W monoblock surface melting is also mandatory and feeds directly into the choice of component shaping. The need for ELM control is strengthened by the switch to full-W, to avoid both melting and excessive W ingress. Studies thus far indicate that melt splashing under disruption transients is unlikely on ITER, and that plasma vapour shielding will significantly reduce the incident transient heat fluxes.

The paper will describe how the IO has constructed, and continues to consolidate the physics basis for the full-W divertor, highlighting areas in which the R&D Community, notably through the International Tokamak Physics Activity, has responded rapidly to requests for input in key areas. It will also present the main features of the divertor design.

This review examines the relationship between tokamak overall performance and the divertor and SOL plasma requirements for handling power and particle flux to material surfaces. The optimization of divertor design must satisfy two different constraints: heat and particle flux at the divertor target consistent with material limitations, and plasma conditions at the boundary core plasma interface consistent with high core plasma confinement and overall performance. Recent results have illustrated the challenge of simultaneously satisfying these two constraints. Devices with high-Z divertor targets (JET, ASDEX-Upgrade and Alcator C-mod) have required low temperature, high density divertor operation to limit the target plate heat flux and accumulation of highly radiative impurities in the core plasma. However, such high density divertor plasmas, and particularly detached divertor conditions have typically been associated with degraded core plasma confinement and overall performance.

Core plasma confinement with its stiff profiles is largely regulated through the edge H-mode pedestal pressure, which in turn is constrained by divertor and SOL plasma conditions and impurities. The parameters at the separatrix that have been observed to affect and potentially degrade pedestal performance include plasma density and temperature, neutral flux, impurity density and plasma turbulence. While existing experiments exhibit pedestal and core confinement degradation if one or more of these parameters exceed some threshold, the scaling of these limits must be fully characterized, and understood, before divertor and SOL solutions for future reactor-scale tokamaks can be designed and optimized. The implications of existing empirical scalings for future tokamak divertors will be explored.

Several other factors for core plasma performance that must be incorporated into divertor and SOL design will also be discussed, including limits to core plasma impurity contamination and radiation, amelioration of ELM transients, and toroidal field volume of the divertor and SOL plasma.

This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences, using the DIII-D National Fusion Facility, a DOE Office of Science user facility, under Award DE-FC02-04ER54698
In this contribution, impacts of the edge stochastic boundary layers on the divertor optimization are discussed. The topics to be addressed are as follows:

1) Effects on a divertor density regime:
   Due to the modification of plasma flow structure and energy transport caused by the formation of edge stochastic boundaries, a divertor density regime can be affected due to a loss of pressure conservation along flux tubes, a replacement of // conduction energy flux with an enhanced \( \perp \) -transport and/or a convective transport.

2) Effects on an impurity transport:
   The magnetic field line braiding can enhance radially outward particle flux. This can lead to an effective impurity screening via a friction force exerted by a plasma flow onto impurities. A modification of edge plasma profiles caused by the stochastization, together with the change of divertor density regime, can affect production of impurity at plasma facing components, too.

3) Effects on radiative layer control:
   In several devices, it is observed that the change of edge magnetic field structure can affect a pattern of radiation layer and thus control of radiative divertor operations. The effects seem to depend on the size and location of an edge remnant magnetic island.

4) Effects on edge electric field formation and turbulent transport:
   It is found that an application of MP changes the profiles of electric field and turbulent transport in the edge region. This has impacts on all the issues listed above and thus is important for understandings of the transport processes.

These effects depend on the edge plasma parameters and magnetic field structures. This contribution attempts to find out possible optimization directions of divertor configurations in future devices taking into account these effects.
The main challenge in realising a fusion power plant (FPP) is the adequate control of heat exhaust fitting to the thermomechanical capabilities of the divertor plasma facing components under the condition of strong neutron irradiation. Already demanding for ITER, the problem is amplified for a FPP where the assumed linear dimensions are about 50% larger and the fusion power output at least 3 times higher. The power crossing the magnetic separatrix is channelled along the magnetic field line to the divertor where it is exhausted on actively cooled divertor targets. The heat flows in a narrow radial layer (SOL) of width $\lambda_q$ (estimated to be ~ 1 mm at the midplane for ITER) scaling only weekly with machine size. This means that the loaded divertor area only scales with the major radius $R$ of a device, making $P/R$ ($P$ being the exhaust power) a crucial parameter when extrapolating to larger devices. For ITER the design requirements for the divertor PFCs are set to 10 MWm$^{-2}$ in steady state (20 MWm$^{-2}$ for slow transients) which is thought to be consistent with the exhaust scenarios tested in present day tokamaks and is already reached for actively cooled tungsten divertor component mock-ups. Although it seems that for these steady state heat loads a solution is at hand, the challenges arise from transient power loads caused by frequent ELMs or sporadic disruptions. For future FPPs the situation is much less settled. Whereas on the one hand the $P/R$ is at least a factor of 2-3 higher, the expected power handling capabilities of the divertor PFCs seem to be rather lower due to the degradation of the thermomechanical performance (mostly embrittlement) of the armour, interface bonding and heat sink materials. On the plasma’s side, scenarios with bulk radiation have to be developed agreeing with the requirement that the fusion performance should not be degraded too much and that the power flowing across the pedestal should be still large enough to keep the plasma in H-Mode. On the materials’ and components’ side new approaches have to be investigated which either will minimize the effect of the neutron irradiation (operation at higher temperature) or will employ newly developed materials which show less degradation of their properties. On top of all these requirements it seems that strong transients, specifically disruption will not be allowed at all.
**R-5: Introduction of HL-2M divertor design**

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HL-2M is a tokamak device that is under construction and put into operation in the near future. The divertor design work of HL-2M carried out at SWIP will be presented. Based on the magnetic coil design of HL-2M, standard divertor, snowflake divertor and tripod divertor configurations are designed. The potential properties of snowflake divertor configurations are analyzed, such as low poloidal field (PF) area around X-point, connection length, magnetic field shear and linear peeling-balloonning (P-B) mode. According to divertor configurations properties of HL-2M, target plates are designed to be compatible with these configurations and to match the requirements of physics and engineering. SOLPS5.0 is used to predict the details of the divertor plasma in these divertor configurations. The constant cross field transport coefficients used in the simulation are estimated according to the heat flux width at outer mid-plane estimated by Eich scaling law. The results show that advanced divertor will dramatically reduce the peak heat flux at outer target by increasing connection length and plasma wetted area. When 10MW heating power flows into the scrape-off layer (SOL) and divertor regions, the peak heat flux at target is less than 2MW/m² with carbon sputter is considered. Owing to the open divertor structure design of HL-2M for advanced divertor operation, it may cause more impurity transports into the core region. From the preliminary divertor design and analysis, HL-2M divertor will satisfy the high performance plasma operation requirements with higher heating power, and will be expected to study advanced divertor physics relevant to next fusion reactor.
R-6: A review on the current status of power and particle exhaust physics: Modeling, experiment and open issues

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Max-Planck-Institut für Plasmaphysik, Boltzmannstr. 2, 85748 Garching bei München

Studying the power and particle exhaust physics remains a key issue when aiming at extrapolating the findings from existing tokamaks to future burning plasma devices such as ITER and DEMO. Physics models implemented into complex numerical codes as well as simple scaling laws are being used for such extrapolations. The scaling laws available are based on experimental data or on numerical simulations. They have only limited range of validity and insufficient sets of such laws currently exist that could be used by system codes for a conceptual design of a future device. On the other hand complex numerical codes have made remarkable progress in recent years in describing the physics processes at work in low recycling as well as for the completely detached divertor. Nevertheless, they fail in numerically reproducing quantitatively as well as qualitatively many experimental observations in the very important transitional regime ranging from high recycling to partial detachment.

This contribution will review the progress made in recent years in validating the numerical code packages against experimental data as well as the developed scaling laws. The shortcoming of the numerical validation will be presented and discussed. The comparison of numerical simulations to experimental data from extensive diagnostics observing the divertor volume allowed the identification of open issues in our understanding of power and particle exhaust physics. Unresolved items that have arisen during these investigations will be presented for discussion. Key issues are perpendicular transport in the divertor, the role of drift terms, the role of CX process for pressure loss and perpendicular redistribution of pressure, the quantification of the volumetric recombination as well as the interaction of power loss with momentum loss processes.

An outlook on activities for the near term future will be given.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
R-7: thinking outside the box: New integrated approaches are needed to solve divertor, main chamber and steady state sustainment challenges for fusion

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The tokamak concept is the most advanced magnetic confinement approach for fusion. But before it can be considered viable for a steady-state, electricity producing power plant, robust solutions for plasma-material interaction (PMI) challenges must be found: (1) power exhaust – order of magnitude improvements in steady state divertor power density handling are needed; (2) divertor target lifetime – target plate erosion must be almost completely suppressed; (3) main-chamber component lifetime – wall components must survive the PMI onslaught for sufficient time, including current drive and heating actuators, which must also attain high overall system efficiency and availability.

In order to meet these challenges – and also satisfy constraints imposed by the DT fusion environment (tritium breeding, magnet shielding/lifetime) – integrated tokamak reactor designs are needed that can fully implement promising advanced divertor approaches and also exploit key attributes of divertor and scrape-off (SOL) plasma physics:

- **Employ double-null topology.** Take advantage of the tendency for heat/particles to exhaust near the outer midplane, preferentially sending fluxes to outer divertor legs.

- **Employ advanced outer divertor legs.** Locate target plates at large major radii (e.g., super-X and X-point target divertor ideas) to decrease target plate heat flux density and provide detachment front stability/control; employ X-points as virtual targets in the divertor volume to intercept peak heat fluxes and increase field line lengths; employ tight neutral baffling for gas-dynamic control; employ liquid metal target concepts.

- **Take advantage of ‘quiescent’ high-field side (HFS) SOL.** Exploit excellent impurity screening properties and profile narrowness (facilitated by double-null) of HFS SOL – locate all close-fitting wall surfaces and RF actuators to the HFS for reduced PMI, reduced impurity sources and core contamination, and highly favorable RF wave physics.

While some of these requirements may appear inconsistent with reactor designs that use current technologies, one should keep in mind that significant advances can occur over a 10-20 year time frame. For example, since the ITER EDA, high temperature superconductors (HTS) and 3D printing of structural materials are now commercially available. Demountable, HTS toroidal field magnets may be possible. This may allow the entire vacuum vessel/divertor of a compact reactor to be replaced as a single unit, as considered in the ARC pilot reactor concept [1]. Such a design might employ: (a) frequent (~1.5 year) replacement of the first wall and actuator components, as well as internal poloidal field coils (perhaps non-superconducting) needed to produce advanced magnetic divertor topologies; (b) use of a full immersion blanket design for enhanced tritium breeding and effective neutron shielding.

Thus, near-term fusion research should focus on exploring/developing/demonstrating physics solutions for divertor and main chamber challenges, without being overly concerned about technology constraints. At the same time, innovative reactor designs using the latest technologies should be investigated.

Finally, to expedite fusion development, a strategy of first targeting a high-field, compact, tokamak that demonstrates net electricity production, even for just a short period of time, may be most effective. This would ‘ignite’ the world’s interest and marshal resources needed to take fusion to the next step – the scale up to a demonstration power plant.

Magnetic confinement devices beyond ITER will face new challenges in plasma exhaust for various reasons, including the higher power handling, increased neutron fluxes and fluence, limitations on measurements and control, higher reliability of components (maintenance issues). Moreover there will need to be a very high level of confidence in solutions to be adopted, given the investments involved. The starting point is to assume that a solution based on that developed for ITER can be transferred to the DEMO scale, and this is the approach used in the present DEMO activity in the EU and some other parties. On the other hand the scale of the challenge for DEMO and power plants means that it is prudent to explore other options, and possibly challenge some traditional constraints and approaches, thereby opening up space for new ideas.

The higher power leads to well-known challenges in terms of power handling in the divertor and the main chamber, and it is both necessary and advantageous to consider the exhaust as an integrated problem, including the main plasma scenario, the scrape-off and divertor plasma, and the engineering, materials and technology. Furthermore the exhaust has to be considered for the whole pulse cycle: start-up, flat-top, ramp-down. If the problem is viewed globally there are both opportunities and impacts relating to systems not traditionally considered in plasma exhaust. There is thus scope for innovative approaches in several areas, especially, but not only, the divertor configuration and plasma-facing components, which might lead to unexpected options. For example the limited understanding of the exhaust plasma is driving new studies (on transport processes and detachment physics for example), and these may reveal ways to increase cross-field transport or create more robust radiative scenarios, and thus alleviate the problem.

This overview will lay out the main issues and some of the possible ways to address them, taking a high level approach.

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I-1: Edge Plasma and Divertor Issues in DEMO-FNS Project

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Conceptual design of divertor and first wall (FW) of DEMO-FNS tokamak being developed in Russia for demonstration of steady state (SS) and hybrid technologies will be presented. DEMO-FNS is a superconducting tokamak of JET/JT-60SA scale (R/a = 2.75 m/1 m). It will operate with the plasma heating power of 40 MW that is close to auxiliary power in these devices. Realization of SS operation in DEMO-FNS accounting for the neutron environment requires new approaches for edge plasma control and divertor/wall operation. Major design challenges are: i) high heat loads onto targets and the FW; ii) erosion of wall and targets due to plasma flows including fast particles; iii) additional erosion under transients (ELMs, VDE, disruptions); iv) damages due to neutron irradiation; v) maintenance, remote handling and integration with other technological systems; vi) necessity of a thin FW for improved neutron balance; vii) lifetime and safety issues. To mitigate high target loads we develop several approaches. Current design of the divertor is based on double null, long outer leg with V-shaped corner that is favorable for plasma detachment. The divertor and FW are water-cooled using CuCrZr as a heat-sink material. To distribute SOL plasma flows between the FW and divertor, the gap is optimized between FW and separatrix (2-3 cm in the equatorial plane). The increased transverse transport due to the narrow gap and non-coronal radiation of impurities (Li, Ne) in SOL and mantle are responsible for splitting heat and particle flows between FW and divertor. Modelling of thermal loads showed that the heat flux density does not exceed 1 MW/m² on FW and 10 MW/m² on divertor targets even in “high-recycling” regime. According to simulations by B2SOLPS5.2 and two-point model, the injection of neon gas in vicinity of targets provides regimes of partial detachment. Lithium dust or liquid jet injection into plasma is addressed in the project. Estimations of lithium and beryllium surface layers under SS operation and transients will be presented together with issues of tritium retention. Neutron damage of PFC in DEMO-FNS will reach ~20 dpa during the device lifetime. Structural/functional materials CuCrZr/Be/SS keep their properties up to 5/10/30 dpa correspondingly. Since the 5 dpa limit restricts the operation of heat sinks to 5 years with 1/3 duty factor, a replacement of CuCrZr elements will be needed. Mockups of the FW and divertor targets were tested at the Tsefey-M facility for 10 MW/m² heat loading.
Interaction of the plasma with neutral gas in the divertor is a most important mechanism opening the way to control of the operational regime of a tokamak reactor. It affects virtually all aspects of the divertor functionality (power loading of the targets, pumping and fuelling, sustaining the operational conditions of the core plasma). In the course of the ITER design development, this interaction has been the subject of intense modelling analysis, supported by experiments on various tokamaks. Neutral gas puffing is found to be the most effective means of divertor control. The results of those studies are summarized and assessed in the paper. The role of the “dome” in the divertor operation, the effect of gas leakage through the targets and other divertor structures, the importance of the neutral particle exchange between the inner and outer divertors and the role of neutrals in the discharge collapse in the case of over-fuelling are discussed. In particular, the massive influx of neutrals into the divertor plasma from the private flux region causes a progressive relaxation of the power flux from the x-point toward the target in such a way that the relative reduction of the power flux density is stronger where the flux is higher. This results in further widening of the power deposition profile and renders the peak power loading of the targets acceptable even for the narrow power SOL upstream (λq ~ 1 mm in ITER) projected by extrapolation of the data from present experiments.
I-3: Key Issues and Goals for Divertor Detachment Performance and Control

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Divertor detachment is central to reducing heat loads to, and erosion of divertor surfaces. Current and past feedback control of the detachment front location using impurity seeding has been very successful at keeping the front either near the target [1] or at the x-point [2], but not in between – essentially an ‘on-off’ switch for detachment location control. That narrow ‘detachment window’ in seeding gas rate corresponding to the location of the front varying from target to x-point, is limiting for ITER and reactors: 1) We will want to control the detachment front location to points in between target and x-point (e.g. the ITER solution) to maximize core confinement, divertor power removal and minimize core impurity levels; and 2) The physics models utilized for detachment control needs to be more general, simultaneously handling other actuators beyond gas seeding (e.g. P_SOL, upstream density…). The narrow detachment window in various actuators itself is a limitation when transients occur; Ideally the divertor can passively handle uncontrolled swings in various core plasma characteristics. We note that unconventional divertor configurations (e.g. snowflake, x-divertor and super-x) may allow better control of the detachment front (e.g. [3]) – beyond their potential to improve the power removal capability of the divertor.

In this talk we will review the current status of divertor detachment with a view towards the implications for ITER and reactors, as described above. We will also present the results of using a modified analytic model [4] of the detachment front location to determine the detachment window in several control variables. We also define a sensitivity of the front location, z, to a given control variable, C, as dz/dC and explore its dependence on front location. Not only are the detachment windows in upstream density, impurity fraction and power into the SOL different, but those windows can expand strongly as the total field at the target, B_t, is reduced compared to that at the x-point, B_x, total (~ the same as toroidal) flux expansion. The implications for conventional and unconventional divertors will be discussed.

References
Detached or semi-detached diverter operation is the primary operational scenario for the ITER diverter (we define a diverter detachment as a "roll over" of plasma flux which in experiment is often observed with increasing plasma density). However, surprisingly so far there is no consensus on the physics mechanism(s) of detachment. To this moment, a few plausible mechanisms of plasma detachment were suggested.

In Ref. 1 the reduction of plasma flux to the target was attributed to the plasma momentum loss due to plasma-neutral coupling (friction). As a result of this friction force plasma flow to the target is slowing down, which could cause the reduction of plasma flux to the target.

In Ref. 2 the detachment was explained by the increase of energy loss (associated with both impurity and neutrals), which reduces the energy flux available for neutral ionization, and/or by the onset of plasma recombination, which literally extinguishing the plasma in front of the target simultaneously reducing the plasma flux to the target and spreading the heat load to the targets by the fast cross-field neutral transport.

Finally, in Ref. 3 it was demonstrated that, for the SOL width comparable and smaller than the ion poloidal gyro-radius, the interplay of the ExB drifts and the parallel plasma dynamics in the plasma momentum balance equation can produce a strong variation of plasma pressure along the magnetic field lines and, therefore, alter plasma fluxes to the targets.

We note that detached regime can be unstable [4] causing a jump of detachment front from the target to X-point following the MARFE formation, which, potentially, can cause disruption of the discharge.

Here we discuss basic physics of diverter detachment and detachment stability. Based on available scalings of the width of the heat flux to the target we derive the scalings for the onset of detachment.

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In-vessel dusts will cause several safety and operational issues for next generation fusion machines like ITER. Several safety limits such as in-vessel dust total quantity have been set within the framework of the ITER project in order to manage the potential dust hazards. This paper deals with the safety issues caused by dusts in tokamaks. The first issue is radioactivity. In future fusion machines, dusts will be activated, tritiated and potentially chemically toxic (presence of beryllium). It is important to estimate the amount of radiation from tritiated dusts. The second issue is tritium retention in dusts. Dusts in tokamaks have irregular shapes and fractal structures. These enhance the specific surface area which leads to much higher tritium retention rate than ordinary surfaces. The third issue is dust combustion. Interaction between hot dusts with water in the case of a water leak will produce hydrogen which can lead to hydrogen or dust explosions. The fourth issue is impact of hypervelocity dusts on PFCs and diagnostics. Hypervelocity dusts are found in several tokamaks. Those dusts can damage tokamak first wall as well as diagnostics installed inside the vacuum vessel.

In this paper, we review the safety issues the dusts in tokamak make, and discuss possible solutions for the problem.
The divertor is a key component of all the modern tokamak. Indeed, it is the most important part of the inner wall which protects the vacuum vessel, and it is the component the closest to the plasma. The most advanced concept for the divertor elements is the water cooled W monoblock concept, which has been chosen for the ITER divertor. It consists in blocks of W joined on a Copper alloy tube. The limits of this concept have been tested at lab-scale on different facilities, and the expected performance is about 10-20MW/m². Such components will be installed and tested in the WEST device to, on a first hand, bridge the gap toward a large scale industrial manufacturing and on a second hand, to operate with such components in a tokamak environment under combined heat and particle loads. A second divertor will also be assembled on WEST, made of Copper alloy elements with a thin W coating (8MW/m² performance). One last concept that is also developed is the W flat tile (10MW/m² performance).

Uncertainties in terms of fabrication are still pending, concerning for example the bonding of the Copper alloy to the cooling system made of stainless steel. These concepts, and in particular the W monoblock, seems reach the technological limits and there is no evident margin of progression to accept higher heat fluxes.

For DEMO and future reactors, it should be assessed if the monoblock concept developed for ITER can be applied. In particular the use of Copper alloy may be a limitation, and its replacement by ferritic steel like Eurofer could be envisaged. Those future devices will require also to take into account the numerous constrains of a nuclear device, not only in terms of conception but also inspection, reliability, maintenance and waste disposal.

To conclude, the divertor conception is a multi-system problem, from the physics of plasma that gives the heat load specifications to the manufacturing process and the qualification of the components. In the future, the constraints imposed for a nuclear reactor device class will increase the number of requirements. It is essential to keep in mind this aspect when dealing with the conception of the future divertor.
I-7: PHYSICS BASIS FOR SIMILARITY EXPERIMENTS ON POWER EXHAUST BETWEEN JET AND ASDEX UPGRADE WITH TUNGSTEN DIVERTORS

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For ITER with metal plasma-facing components (PFCs) impurity seeding is essential for power dissipation in the edge/SOL region. Scaling laws for describing the physics of edge plasma transport and power dissipation by interaction with neutrals and radiation are not available. Instead plasma edge codes like SOLPS are utilized to derive operational constraints for which the model assumptions for highly radiative scenarios have to be revisited and validated. Present day tokamak devices differ in geometry and usually the parallel power flux density $q_\parallel$ does not match. A similarity experiment for radiative (seeded) scenarios in JET and ASDEX Upgrade with W divertor and relevant parameters matched however would allow the closest comparison possible for the power dissipation mechanism.

A dedicated similarity study of the power exhaust problem in its strictest sense would allow scaling laws to be derived for transport and the power dissipation mechanism as function of basic edge plasma parameters only, i.e. normalized collisionality $v^*$ and gyro-radius $\rho^*$, $\beta$ and the Debye length $\lambda_D$. Unfortunately, a complete identity experiment for the plasma core and the edge is however not possible when the atomic physics is included [1]. Assuming an “isolated divertor” approach [1] and by restricting the set of similarity parameters to plasma temperature $T$ and the product of upstream density and connection length $n_m L_c$ at the mid-plane (and thus $v^*$ in the SOL), one identifies the power density parameter $P_{sep}/R$ as the main operational control parameter for power exhaust similarity studies. This quantity is directly proportional to $q_\parallel$ flowing in the near SOL (and inversely proportional to the heat flux width $\lambda_q$). In this approach, however, similarity in both $\rho^*$ and $\beta$ is not possible and thus particle drifts and MHD effects are masked in the scaling. A further relaxation of other geometrical parameters like the divertor field pitch angle $\alpha_D$ allows for a similarity in both $\rho^*$ and $\beta$ under certain conditions [2]. The required power for such a matched experiment scales as $P_{sep}\sim R^{1.5}$ and is thus less demanding.

A review is given of the accessible similarity parameters for a scaling experiment on power exhaust for JET and ASDEX Upgrade divertor with W PFCs and radiative dissipation through extrinsic seeding. A set of accessible control parameters is presented which allows for proper scalings for both, the edge transport and the radiation efficiency in reducing $q_\parallel$ in partially detached conditions.

I-8: HIGH RADIATION SCENARIOS IN PRONOUNCED DETACHED DIVERTOR CONDITIONS AT ASDEX UPGRADE

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Future fusion reactors require a safe, steady state divertor operation. A possible solution for the power exhaust challenge is the detached divertor operation in scenarios with high radiated power fractions, increasing the power dissipation to more than 90%. The radiation can be increased by seeding impurities, such as N for dominant scrape-off-layer radiation and Kr for dominant core radiation (as it is required for a device like DEMO).

Recent experiments on the full-tungsten ASDEX Upgrade (AUG) demonstrate the quasi-stable operation with high radiated power fractions ($f_{\text{rad}}\leq90\%$) at high heating powers ($P_{\text{heat}}/R=12$-$13\text{MW/m}$) with a conventional divertor using a vertical target geometry. These scenarios include either strong nitrogen or krypton seeding to increase the radiation and induce detachment. Other seed impurities will be tested for these scenarios in the future. The neutral gas pressure in the divertor is required to be more than 2 Pa in order to mitigate internal magnetic modes (by reducing the plasma $\beta$) and to facilitate detachment.

With both seed impurities, a radiation condensation occurs and the dominant radiation originates from the confined region of the plasma. With Kr seeding, this takes place in the pedestal region and leads to a radially localized radiating ring. In the case of N seeding, the radiation condensation leads to a poloidally localized radiator in the region of the X-point (MARFE-like). Unlike in the operation with carbon as first wall material, the plasma with such an X-point radiator remains stable. Recent experiments at the JET tokamak show similar X-point radiator patterns for N, Ne and Ar seeding, but appear to be limited to a maximum radiated fraction of 75% for stable operation [1]. The aim is to achieve a better understanding of the high radiative scenarios by comparing the observations at these two all-metal devices.

For both scenarios in AUG, the divertor targets are detached along more than 10 $\lambda_q$ (except during transient events) and an increase of the plasma density is observed. There is a significant reduction of the plasma temperature, however, the impact on the energy confinement time varies; for N seeding, the core pressure is observed to stay constant while the pedestal top pressure reduces, leading to a small reduction of the confinement. For Kr seeding, the impact on the confinement depends on the location of the dominant Kr radiation and can be constant or decreasing.

I-9: THE EUROPEAN R&D PROGRAMME ON DIVERTOR ARMOR MATERIALS AND TECHNOLOGY – STATUS AND STRATEGY


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The design of plasma-facing units is a multi-dimensional task, which – among others – strongly depends on divertor/plasma operation conditions, the choice of coolant, safety and licensing considerations. Unfortunately, for DEMO and power generating fusion reactors, many of the divertor requirements, specifications and parameters are not completely available yet. To bypass this lack of knowledge, current concepts are based on very conservative assumptions in order to mitigate the risk of failure at a later development stage, which in turn might rule out less efficient but still viable concepts. Since material selection is one of the key elements, the present contribution reviews this complex of problems and focuses on the current status and future strategy of the European R&D programme. Important material data for plasma-facing units include mechanical, high heat-flux and thermal properties, corrosion in connection with the coolant, erosion and plasma interaction phenomena. Moreover, reliable designs require the knowledge of the deterioration of all these properties during neutron irradiation. In this context, severe gaps in the materials database are pointed out. Nevertheless, the available experimental and simulation results are discussed with respect to current DEMO operating specifications, material and design limits. Given the current design and materials understanding, the conclusions that can be drawn reveal that no solution for the DEMO divertor yet exists. However, reasonable limits for the allowable heat flux, preferable temperature ranges for various materials, and possible consequences for the component design are now emerging. In addition, these limits allow for a definition of specific objectives in the materials R&D programme and for developing a strategy to reach them. The discussion is illustrated by selected highlights from present research projects.
I-10: Materials for DEMO and Reactor Applications — Boundary Conditions and New Concepts

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For the first wall of a fusion reactor unique challenges on materials in extreme environments require advanced features in areas ranging from mechanical strength to thermal properties. The main challenges include wall lifetime, erosion, fuel management and safety. For the lifetime of the wall material, considerations of thermal fatigue as well as transient heat loading are crucial, since typically 10¹⁰ (30 Hz) thermal events during one year of operation are to be expected. Tungsten (W) is the main candidate material for the first wall of a fusion reactor as it is resistant against erosion, has the highest melting point of any metal and shows rather benign behavior under neutron irradiation. To overcome brittleness issues when using W, a W-fiber enhanced W-composite material (W/W) incorporating extrinsic toughening mechanisms can be used. These mechanisms enable energy dissipation and thus stress peaks can be released at crack tips and cracks can be stopped. Accordingly, even in the brittle regime this material allows for a certain tolerance towards cracking and damage in comparison to conventional tungsten. First W/W samples have been produced, showing extrinsic toughening mechanisms similar to those of ceramic materials [1]. These mechanisms will also help to mitigate effects of operational embrittlement due to neutrons and high operational temperatures. A component based on W/W shall be developed with both chemical infiltration (CVI), utilizing a newly installed CVI-setup and a powder metallurgical path through hot-isostatic-pressing. New manufacturing approaches and specially structured materials e.g. PIM Tungsten may address issues of embrittlement. Addressing the safety issue, a loss-of-coolant accident in a fusion reactor could lead to a temperature rise to 1400 K after ~30–60 days due to neutron induced afterheat of the in-vessel components [2]. Thereby, a potential problem with the use of W in a fusion reactor is the formation of radioactive and highly volatile WO₂ compounds. In order to suppress the release of W-oxide tungsten-based alloys containing self-passivating alloying components seem feasible, as they can be processed to thick protective coatings with reasonable thermal conductivity. Enhanced sputter erosion during normal reactor operation is not expected to be a concern as preferential sputtering of alloying elements leads to rapid depletion of the first atomic layers and leaves a pure W-surface facing the plasma [3]. W-Cr-Y with up to 80 at% of W content already shows 10⁵-fold suppression of tungsten oxidation due to self-passivation. Rigorous testing of oxidation behavior, high heat flux testing and plasma loads as well as mass production for candidate materials will be performed. Developments joining W as PFM with the structural material EUROFER97 via Functionally Graded Materials (FGMs) are ongoing, they can mitigate the effect of mismatch in the thermo-mechanical properties. Furthermore, tritium management remains an issue despite the low tritium retention in W. In order to prevent tritium loss and radiological hazards it is important to suppress permeation through the reactor walls [4]. Permeation barriers require high reduction factors, high thermal stability and corrosion resistance as well as similar thermal expansion coefficients compared to those from the substrate are required for barrier layers. A new deuterium gas-driven permeation setup is used to investigate the deuterium permeation through different ceramic coatings on EUROFER97, which significantly reduce the deuterium permeation [5]. Finally, for the development of plasma facing components the issues of power exhaust needs to be considered. This might require replacing copper as a heat sink e.g. by steel to avoid irradiation-induced deterioration, e.g. swelling [6]. However, this deteriorates thermal properties and worsens the activation behavior as is also seen when using interface materials such as oxides in composites and as permeation barriers [7].

To mitigate the risk that the conventional divertor solution adopted for ITER, which is based on a single-null magnetic configuration and tungsten (W) targets, will not extrapolate to a fusion power plant, the European fusion consortium (EUROfusion) is assessing alternative divertor solutions. Considered alternatives include non-conventional configurations and the use of liquid metals as divertor armour. Their performance and costs is compared to those of a conservative DEMO design targeting 500 MW of electric power with a major radius of 9 m, a toroidal field of 5.8 T and a plasma current of 20 MA.

The assessment of alternative configurations considers the snowflake divertor (SFD), the X divertor (XD) and the Super-X divertor (SXD). Potentially higher costs such as a higher poloidal field (PF) coil and toroidal field (TF) coil volume, higher forces on PF coils and a higher fraction of neutrons lost for breeding are weighed against predictions of the performance improvement with regard to access to and stability of operation in the detached regime and any increase of the power that can be radiated in the divertor using state-of-the-art boundary models. The models and, in particular, their dependence on magnetic geometry will be tested with experiments planned in TCV and MAST-upgrade as well as in collaboration with other international devices. The assessment of liquid metal based divertor solutions focuses on a heat conduction based capillary porous system (CPS), which is viewed as the least complex liquid metal based solution. While the heat exhaust potential of such a solution faces similar challenges as W targets, key advantages are the in-situ repair of eroded material and potentially a greater tolerance for transient heat flux excursions. Materials considered are lithium (Li), tin (Sn) or Li/Sn alloys. It is, in particular, planned to demonstrate the stationary operation of a CPS solution including cooling and replenishment in the FTU tokamak.

The assessment will provide input to refine the scope and the technical specifications of a dedicated European Divertor Tokamak Test (DTT) facility.
I-12: REACTOR DIVERTOR DESIGNS BASED ON LIQUID METAL CONCEPTS

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The use of liquid metals as plasma facing components in Fusion devices was proposed as early as 1970 for a field reversed concept. The idea was extensively developed during the APEX Project, in the turn of the century, and it is the subject at present of the biannual International Symposium on Lithium Applications (ISLA), whose fourth edition will take place in Granada, Spain at the end of September. While liquid metal flowing concepts were specially addressed in the USA research projects, the idea of embedding the metal in a capillary porous system was put forwards by the Russian teams in the 90’s, thus opening the possibility of static concepts. Since then, many ideas and accompanying experimental tests in fusion devices and laboratories have been produced, involving a large fraction of the countries within the International Fusion Community. Within the EuroFusion Road map, these activities are encompassed into the Working Programs of the PFC and DTT packages.

In this presentation, a review of the existing liquid metal-based divertor alternatives for a Fusion Reactor and their degree of development will be presented. In particular, the conclusions drawn from the Panel Discussion to be held during the ISLA event, almost concomitant to this Technical Meeting, will be addressed and discussed.
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| O-2  | J. HARRISON                        | Implications of Filamentary Transport in the Divertor for Exhaust Design |
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| O-8  | G. CIRAOLO                         | Impact of magnetic and PFC geometries on neutral control: SolEdge2D modelling comparing WEST and ASDEX-U plugging performance |
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| O-10 | H. YOU                             | Critical issues & challenges in the engineering of DEMO divertor target |
| O-11 | B. XIAO                            | Study of the advance divertor magnetic configuration on EAST under steady-state condition |
| O-12 | R. GOLDSTON                        | The Lithium Vapor Box Divertor |
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| O-14 | V. SOUHANOVSKII                    | Developing Snowflake Divertor Physics Basis in the DIII-D, NSTX and NSTX-U Tokamaks Aimed at the Divertor Power Exhaust Solution |
| O-15 | B. LABIT                           | Experimental Studies of the Snowflake Divertor in TCV |
| O-16 | G. MAZZITELLI                      | Experimental results with the Cooled Lithium Limiter (CLL) on FTU |
| O-17 | R. ALBANESE                        | DTT: a Divertor Tokamak Test facility for the study of the power exhaust issues in view of DEMO |
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Concepts for Fast Flowing Liquid Lithium Walls and Divertors

O-19 M. Ye
Study of Quasi-Snowflake Divertor for Cfer by using Solps

O-20 L. Wang
ITER-like tungsten divertor development and experiments on EAST
O-1: CONTRIBUTION OF JT-60SA TO POWER EXHAUST STUDIES IN VIEW OF ITER AND DEMO

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The JT-60SA tokamak with superconducting coils is currently under construction and is scheduled to begin operation in 2019. The device is designed for integrated ITER and DEMO relevant plasma research including pulses longer than 100s. This contribution will present the envisaged evolution of the divertor and of the plasma facing components, PFCs, in view of the changing scope of the device. JT-60SA will be well equipped with diagnostics relevant for power exhaust studies. Operation of the divertor will start with bolted CFC tiles and water-cooled mono blocks with C as PFC. For long pulse operation at high beta fully water-cooled C mono blocks are foreseen. For current drive experiments at low density the heat load is foreseen to be mitigated by Ar seeding according to simulations using the SONIC code package. As a follow up and in order to contribute to the ITER and DEMO relevant integrated power exhaust studies the potential for changing the plasma facing components to metallic PFCs is currently being examined. The principal corner stones for such a change over are going to be presented.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
Studies of filamentary transport in the divertor of MAST using high-speed imaging of D$_\alpha$ light have improved our understanding of particle transport in the scrape-off layer (SOL) and private flux region (PFR). Filaments appear in three regions in the divertor, 1) in the PFR, 2) in the SOL close to the equilibrium separatrix and 3) deeper into the SOL away from the separatrix in both L-mode and H-mode. The size, velocity and lifetime of these filaments are found to vary significantly from one region to the other, suggesting distinct filamentary transport mechanisms. The lifetime of filaments in region 3 is <8μs in region 2, and ~100μs in regions 1 and 3, comparable with the parallel ion transport time. Preliminary 2D simulations of the filaments in region 1) have been modelled with the BOUT++ code, and have been able to qualitatively reproduce their appearance and propagation. An experimental characterization of the filaments in the private flux region will be summarised alongside a comparison of the filaments in the divertor with the decay lengths of ion saturation current profiles in the PFR inferred from Langmuir probes. It was found that the filaments in the PFR appear brightest on the inner leg, thought to be due to the higher neutral density in this region, and move poloidally toward the inner target. Furthermore, the characteristics (size, velocity and lifetime) of the two categories of filaments in the SOL (regions 2 and 3) will be presented and compared with the radial decay length of the ion flux to the divertor to estimate the role of these filaments in establishing the shape of the profiles.

These findings will be used to suggest experiments where filamentary transport in the divertor can be modified through changes to the magnetic geometry of the divertor leg, thereby changing the alignment of the radial pressure gradient and the field line curvature. The aim of these experiments would be to broaden the SOL by increasing cross-field transport, which could be a consideration in the design of the divertor for DEMO.

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A global understanding of turbulent cross-field transport is mandatory to address the properties of plasma-wall interaction, including the edge plasma, the Scrape-Off Layer and the divertor plasma. Turbulent transport governs the SOL width, the large-scale flows and electric potential structures, all the so-called drift effects.

Turbulence modelling is challenging due to multi-scale physics and inhomogeneous three-dimensional features. This includes in particular the micro-current pattern that controls turbulence. To be predictive, simulations must address complex magnetic configurations, obviously the X-point, and the plasma shaping that plays a role in the edge confinement properties.

Turbulent fluid simulations are performed at the plasma edge using the first principle 3D code TOKAM3X. First simulations have been run in divertor configurations, using JET-like and COMPASS-like magnetic equilibria.

Turbulence characteristics and their spatial distribution show common features with former simulations and experimental observations in limited plasmas; large poloidal asymmetries in radial turbulent flux as well as positive skewness of the PDFs in the SOL are observed. Good qualitative agreement with experimental results has been also found for the parallel Mach number profile. The stagnation point is found to stand between the low field side midplane and the X-point, while the top Mach number lies in the range 0.3-0.4. In addition, large asymmetries in the poloidal direction have been found for the density decay length.

Sensitivity to separatrix geometry is observed, and will be discussed in the frame of divertor concepts.
O-4: SIMULATIONS OF BOUNDARY PLASMA IN X-POINT TARGET DIVERTOR CONFIGURATION

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The enormous challenge of solving the divertor problem for a tokamak-based fusion reactor prompts the search for innovative solutions, and in recent years in the tokamak edge plasma community there has been significant interest in innovative divertor configurations with secondary X-points in the divertor volume or close to the target plate. Such configurations include the cusp divertor [1], snowflake-like divertor [2], X-divertor [3], and X-point target divertor [4]. The presence of a secondary X-point in the divertor has hindered the application of established tokamak edge plasma transport codes such as UEDGE and SOLPS. On the other hand, stellarator edge modeling tools, specifically the EMC3-Eirene package, have more geometric flexibility and can be applied to configurations with two or more X-points; however the plasma model in EMC3 currently lacks cross-field particle drifts [5].

To meet the demands for comprehensive modeling tools for such innovative configurations, the UEDGE code and its grid generator have been recently advanced to allow for inclusion of a secondary X-point in the divertor region. In terms of physics applications, of particular interest for the present study is the X-point target divertor configuration that has been speculated to allow for stable, highly radiating detached plasma regimes in the ADX tokamak concept [4]. Here results of application of the upgraded UEDGE to an X-point target divertor configuration will be presented and physics implications will be discussed.

O-5: Achievements and Challenges in Automated Parameter, Shape and Topology Optimization for Divertor Design

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Divertors need to be designed to handle power exhaust and to provide sufficient Helium pumping capacity. Plasma edge transport codes such as B2-Eirene [1] are indispensable for the design of future divertor concepts. However, long simulation times for analysis, different design requirements and constraints, combined with the large number of control variables turn divertor design into a challenging problem. Over the past decades similar design challenges are faced both in aerodynamics and in structural mechanics. Shape optimization was introduced to design airfoils for drag reduction or lift maximization [2]. Topology optimization was used to design sufficiently strong constructions at minimal weight and material costs [3]. Both problems have been treated very effectively by using optimization approaches including adjoint PDE formalisms for computing design sensitivities.

Recently, adjoint based automated design methods were explored for fusion reactor and heat sink design. Optimal solutions were achieved through gradient-based optimization algorithms, where adjoint sensitivity analysis is used to keep gradient evaluation costs sufficiently low.

In previous work [4], it is demonstrated that shape optimization methods applied to divertor shape design can efficiently propose divertors which optimize target power load spreading. Also radiation heat loads, computed by a Monte Carlo method, could be incorporated within this methodology [5]. A similar, though now in parts adjoint methodology is developed for magnetic configuration design [6]. Again, the optimization methodology successfully alters the magnetic configuration to reduced heat peaks. Further, the perspectives of topology optimization for cooling designs was investigated and led to branch-like cooling channel patterns for micro-channel heat sinks [7]. In the present paper, the status, perspectives and challenges for optimization tools both in divertor configurations and cooling designs will be reviewed.

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The demonstration fusion reactor DEMO is the next step in the European fusion roadmap after ITER, and has as primary aim net electricity production demonstration in a fusion plant. The present European DEMO design [G. Federici FED 89 (2014)] has more challenging physics and engineering constraints in comparison to ITER. One of the major aspects, which motivated this analysis, is that DEMO has more than 3 times the ITER heating power, but has less than 1.5 times major radius. Another DEMO unique requirement is the tritium breeding cycle self-sufficiency, which, in combination with the need to produce electricity, adds a more restrictive technological wall heat load limitations, assessed to be, for the presently developed breeding blanket concepts, up to 1.5MW/m², for the water cooled solution, and 1MW/m² for the He cooled designs [R. Wenninger NF 55 063003 (2015)]. In ITER more than 50% of the first wall is specified for more than 3.5MW/m². Due to this low limit it is important to get a good estimation of the total load estimation on the first wall. This contribution describes first investigations of the most relevant load types. Two of the main sources of PFC heat loads are analysed in this work. The first contribution regards the load transferred by charged thermals. The analysis is performed via 2D field-line calculations on DEMO magnetic equilibria and lead to the results, in the worst case unrealistically pessimistic, of having all the power in the far SOL decay length with a $\lambda_q \sim 6\text{cm}$ and no radiation, of 0.9MW/m² in the upper inboard first wall.

The second analysis regards the radiated heat flux density, reported in [R. Wenninger NF 55 063003 (2015)], for which a peak power flux density during flat-top operation on the wall due to radiation can reach up to 0.45MW/m² for DEMO relevant conditions. A more detailed analysis on how to extrapolate the amount of radiation clustering in the x-point vicinity as a function of the impurity mix is being performed. While the contribution of the charged particle and the radiation presented cannot be added, because the first consider no radiation, while the second a radiation of over 90%, they do not include 3D features of the plasma and the first wall, which would have the peaked values higher than the nominal. These analyses are used at this stage to translate the uncertain physical parameters in engineering wall load specification boundaries.
Tungsten is due to its unique properties a prominent candidate for highly loaded areas in a future fusion reactor. In this respect the development of advanced tungsten materials is essential for sophisticated divertor concepts. Promising candidates are tungsten fibre-reinforced tungsten composites (Wf/W) which utilize extrinsic toughening mechanisms and therefore overcome the intrinsic brittleness of tungsten. Tungsten composites have been successfully produced and tested during the last years and the focus is now put on the technological realisation of Wf/W based divertor targets. Besides the huge particle and energy fluxes the divertor targets will face severe challenges like as an increase of the DBTT due neutron irradiation and potentially strong recrystallization at least during thermal transients further promoting thermal fatigue. In this contribution we critically discuss the aspects of a divertor concept based on Wf/W in respect to these challenges.

As the toughening mechanisms in Wf/W are purely mechanical, toughening is still active after embrittlement and thus the effect of operational embrittlement is mitigated. Potassium doped wire used as fibres in Wf/W show an excellent high temperature performance up to 2000 K and allow therefore an increase of the operation temperature. The good toughness of Wf/W at room temperature allows the extension of the temperature window to lower temperature. Targeted fibre positioning could be used to strengthen and toughen critical parts like regions with stress concentration or high risk of crack initiation. On the other hand, the use of composites raises new aspects within classical divertor materials and plasma wall interaction issues. E.g. composites feature many internal interfaces which might have an impact on thermal conductivity and hydrogen retention. For a successful realisation a strong collaboration of materials and fusion experts as well as design engineers is necessary.
Operating a fusion reactor requires handling high heat loads on the divertor plasma facing components and controlling particle recirculation within the divertor volume. In order to mitigate risks for the ITER divertor procurement and operation, the WEST project aims at transforming Tore Supra into a diverted machine able to study actively cooled tungsten monoblocks under conditions representative of ITER long pulse operation (heat flux, plasma fluence). A considerable simulation effort has been dedicated over the years to properly estimate the power loads on the materials and understand diverted plasma specificities in terms of impurity screening. In this context we report recent results from the SolEdge2D plasma transport code coupled to the Monte-Carlo EIRENE code for neutral particles transport. Complex and realistic geometries can be handled by SolEdge2D thanks to the penalization technique, allowing us to properly take into account the interaction between the plasma and the multiplicity of objects located in the vessel (assuming axi-symmetry). Main chamber recycling and sputtering, which can play a substantial role in today machines, and may be very important in an all-W reactor, can thus be addressed on a sounder basis. More generally, we can address the synergy between plasma transport and geometry of plasma facing components on, for example, neutral particles recirculation or impurity contamination, gaining insights into the influence of divertor, including the secondary X-point effect, and baffle geometries on the plasma edge conditions.

In this contribution we investigate plasma properties for simulated H-mode plasmas on a WEST configuration. In order to set the radial transport coefficients and impose a transport barrier at the edge, we first reproduce H-mode plasma on ASDEX Upgrade, referring to a well document discharge (see A. Chankin et al. PPCF 48, 2006, 839–868). A careful particle balance can then be carried out. The change in recycling pattern with increased density and the small neutral leak from the divertor to the main chamber determine the maximum pulse length before a deleterious transition from divertor recycling to main chamber recycling.
A major challenge facing the design and operation of next-step high-power steady-state fusion devices is to develop a viable divertor solution with order-of-magnitude increases in power relative to present experience, while having acceptable divertor target plate erosion and being compatible with maintaining good core plasma confinement. DIII-D features well-established ITER-like and advanced tokamak scenarios with extensive boundary/divertor diagnostics, aimed at (1) understanding and validating the models of divertor plasma detachment, (2) optimizing the divertor to maximize divertor dissipation at lower upstream densities to facilitate core/edge integration, and (3) developing advanced divertor concepts for future fusion devices. Recently, we have identified a radiation shortfall in the 2D fluid codes, including both UEDGE and SOLPS, which underestimates the radiative losses in the divertor as the plasma detaches. Further efforts are being made to understand atomic/molecular physics that control volumetric power and momentum losses and quantify parallel and perpendicular transport, especially near detachment. Promising progress has been made on DIII-D in developing advanced divertor solutions, including both snowflake divertor (SFD) and X divertor (XD), leveraging DIII-D’s robust plasma control system and great flexibility in the magnetic configuration. To examine the effect of divertor closure, a comparison will be made between the open lower divertor with the more closed upper divertor in DIII-D in the next experimental campaign. Code studies by Asakura et al [1] have identified a promising divertor configuration for JAEA’s SlimCS DEMO involving a narrow slot with a V-shaped corner. We are using SOLPS to model a scaled version of this configuration for tests in DIII-D, planned for 2017. Initial modeling of such further closure of the DIII-D upper divertor indicates that similar beneficial effects should occur in the scaled version, namely a reduction relative to the case of a rectangular slot of divertor temperature and target heat flux by redirecting and confining recycling neutrals and impurities near the V-shaped corner.

Development of a diverter target with a sufficient capability of power exhaust is a crucial prerequisite for the realization of a fusion power plant. While the design and technology for divertor target has been successfully developed for ITER, the applicability of this concept is not necessarily assured yet for DEMO mainly because the neutron irradiation dose expected for DEMO divertor will be by an order of magnitude higher than that of the ITER divertor. The possible embrittlement of structural heat sink materials due to irradiation is likely to restrict the structural performance and the operational flexibility of a target component to a considerable extent. For judgment of design feasibility of a novel target concept a quantitative evaluation of the thermal and structure mechanical performance is needed.

In this contribution, a brief overview is presented on the critical issues of the materials/design interface for the engineering of DEMO divertor target. Emphasis is put on the mutual impact between the design requirements and the performance of structural and armor materials. In addition, a concise review on the two conventional target design concepts is given. Finally, an outlook of novel design concepts and the related technology trend are introduced.
**O-11: STUDY OF THE ADVANCE DIVERTOR MAGNETIC CONFIGURATION ON EAST UNDER STEADY-STATE CONDITION**

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Heat and particle loads on the divertor are among the most challenging issues to be solved for next step fusion reactor [1, 2]. Alternative magnetic configurations may enable tokamak operation with a lower peak heat load than a standard Single Null (SN) divertor. These papers reports on the control of one of such alternatives - a divertor configuration with two close nulls called Quasi-Snowflake (QSF) [3] and the concept of Fish-tailing divertor on EAST. Up to 20 second QSF have been achieved under nearly truly steady state (vp=0) with injection of lower hybrid waves.

The nature of EAST with limited amount of the shaping coils with limited Ampere-turns and the far coil distances from plasma result in less flexibility and more difficulty in snowflake-like plasma shape. In 2014, a QSF shape with the second null distanced significantly from the primary one has been achieved. An increase of the connection length by ~30% and the flux expansion in the outer Strike Point (SP) region by a factor ~4 has been obtained with respect to the conventional EAST SN discharges, confirming the prediction by CREATE-NL tools [4]. It has been observed that in Low-Confinement (L-mode) discharge the peak of ion saturation current density in Langmuir Probes (LPS) drops once the QSF shape becomes stable compared to a SN case that could indicate a heat flux reduction. Preliminary interpretative 2D edge simulations have been performed using the TECXY code [5] showing a good agreement between the Infrared Camera (IR) measured and simulated peak heat load that highlight a reduction for this quantity in QSF case, mainly due to the increase of the flux expansion with respect to the SN. In addition, predictive 2D edge simulations highlighted that the heat flux mitigation apparently improves at highest densities, and should be particularly evident with high additional heating power, since a stronger absolute drop of the loads has to develop for the same mitigation factor.

The second option for the heat spread is to use fish-tailing divertor. It is to reduce peak heat flux by fast scan of the strike points in certain amount of the divertor target area by set up a coil located near the divertor target and apply fast AC current. The initial calculation confirmed that if there is 5KA AC current on a coil with distance of 10cm from the divertor target, heat load could be spread out in a width of 10 cm in a uniform distribution. This would result in the reduction of peak heat flux from 10 MW/m\(^2\) to 2MW/m\(^2\). The optimization of this coil location and current is in progress.

In 2015, the shape feedback control has been implemented for the control of the snowflake shape control. Heat flux expansion and higher particle removing rate have been demonstrated. 20-second plasma discharges has been achieved with the stable control of the quasi-snowflake shape.

**References**

O-12: THE LITHIUM VAPOR BOX DIVERTOR

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It has long been recognized that volumetric capture of the plasma efflux from a fusion system is preferable to its localized impingement on a material surface, in order to mitigate the anticipated both very high heat flux and intense particle-induced damage. This is the fundamental motivation behind the “gas-box” divertor concept\textsuperscript{1}, in which recycling DT fuel is to provide momentum balance with the upstream plasma, through charge-exchange and collisional friction, allowing the divertor plasma to detach from the material target. Full detachment, however, generally results in the high-neutral-density detachment region moving to the magnetic x-point\textsuperscript{2}, and in deterioration of plasma confinement and increased impurity levels. Projections to a demonstration power plant\textsuperscript{3}, furthermore, suggest an immense upstream parallel heat flux, of order 12 GW/m\textsuperscript{2}, so fully detached operation may be a requirement for the success of fusion power. Building on earlier work by Nagayama et al.\textsuperscript{4} and by Ono et al.\textsuperscript{5}, we present here a concept for a lithium vapor box divertor. Our approach can be viewed as a combination of the two earlier ideas. We propose to use a series of differentially pumped “vapor boxes” to isolate a high pressure of lithium vapor from the main plasma chamber and so control the location of plasma detachment. Such powerful differential pumping is only available for condensable vapors, not hydrogenic gasses. We demonstrate the properties of such a system through conservation laws for vapor density and enthalpy, and then include plasma entrainment and ultimately a conservative estimate of radiated power. In the extreme case, the lithium can provide pressure balance for full detachment from the main plasma. Alternatively, a lower vapor pressure can radiate enough power from the plasma that it should recombine and then pressure balance would be achieved between the flowing lithium plasma and its recombined vapor. This approach appears attractive, but further analysis and experimentation will be required to demonstrate its practicality.

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\textsuperscript{4}Y. Nagayama et al., Fusion Eng. Des. \textbf{84} (2009) 1380
\textsuperscript{5}M. Ono, M.A. Jaworski, R. Kaita et al., Nuc. Fusion \textbf{53} (2013) 113030
O-13: Lithium and Liquid Metal Studies at PPPL

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Strategic planning studies have identified control of the plasma-material interface as a critical area for realization of power production. Solid plasma-facing components (PFCs) are the leading candidates for future devices, predominantly serving as PFCs for present devices. ITER is relying on metallic PFCs, namely W in the divertor and Be on the first wall. While ITER has been designed to work with these materials, there is little safety margin on heat flux removal capability. The power exhaust challenge for reactors the size of ITER is even harder, requiring higher amounts of core and divertor radiation. Moreover, very recent studies have shown that both the steady heat exhaust and transient exhaust, during e.g. edge-localized modes (ELMs), is more challenging, owing in part to the narrowness of the scrape-off layer power flux footprint with increasing midplane poloidal field.

Liquid metal (LM) PFCs have some attractive features that could remove some of the restrictions of solid PFCs. The typical erosion and PFC performance degradation of solid PFCs can be obviated with self-healing surfaces; the challenge shifts to controlling core impurity content, and managing tritium retention. Similarly LM PFCs are also tolerant to neutron damage, and can, under the right conditions, exhaust very high steady and transient heat flux. Finally lithium PFCs can provide access to low recycling, high confinement regimes, e.g. at ~2x H-mode scalings, enabling attractive core and edge plasma scenarios. Outstanding issues include acceptable temperature operating windows, LM chemistry control, and free-surface MHD issues for unrestrained flowing LM systems.

This paper focuses on research and development of liquid metal and lithium PFCs at PPPL, including NSTX and coming studies in NSTX-U, along with studies in smaller devices, e.g. LTX, as well as collaborative studies on other US and international devices; a related paper at this workshop discusses the vapor box divertor concept. The use of lithium often increased the energy confinement in these devices, and modified the edge characteristics. An overview of experimental results and R&D directions will be presented. Supported in part by US DOE Contract No. DE-AC02-09CH11466.

Menard J. E. et al., Nucl. Fusion 52, 083015 (2012).
O-14: Developing Snowflake Divertor Physics Basis in the DIII-D, NSTX and NSTX-U Tokamaks Aimed at the Divertor Power Exhaust Solution.


The snowflake (SF) divertor magnetic configuration [1] enables significant manipulation of divertor heat transport for power spreading and reduction in attached and radiative divertor conditions, between and during edge localized modes (ELMs), while maintaining good H-mode confinement ($H_{98y2}>1$), as recent NSTX and DIII-D studies show [2,3]. These results, in combination with reactor relevant magnetic coil layout and SF equilibria studies [4,5], suggest the SF divertor is a promising tokamak power exhaust concept.

The present vision for controlling the plasma–material interface of a tokamak is an axisymmetric poloidal magnetic X-point divertor. The ITER divertor is based on the standard geometry tested in tokamak experiments and uses tilted vertical targets to generate partial radiative detachment of the strike points. However, the standard radiative divertor approach is likely to be insufficient for DEMO-like devices. Alternative (w.r.t. the standard divertor) magnetic divertor configurations have been an active area in fusion plasma research since 1970s. The tokamak concepts included long legged divertors, expanded boundaries, multiple X-point divertors, and multi-pole divertors [6]. These concepts have a potential to modify steady-state and transient particle and power exhaust via modifications to parallel and perpendicular transport, increased dissipative loss channels and plasma-wetted area.

In recent years, the snowflake (2nd order null) divertor configuration was implemented in the TCV, NSTX, DIII-D and EAST tokamaks with existing magnetic coils. The DIII-D results include: 1) Increased scrape-off layer (SOL) width that may imply enhanced radial heat transport; 2) Significant peak inter-ELM and ELM heat flux reduction, due to modified geometry, higher radiation, and heat and particle flux sharing over additional strike points; 3) In radiative D2-seeded SF divertor, nearly complete inter-ELM power detachment at $P_{SOL}$$\sim$4 MW, with broader divertor radiated power distribution (cf. standard divertor), albeit with some confinement degradation; 4) First measurements of divertor null-region poloidal beta $\beta_p$$>>1$, consistent with the fast convective plasma mixing mechanism [7]. In NSTX, the SF divertor led to a stable partial divertor detachment with inter-ELM peak heat flux reduction by factors 3–5, and peak ELM heat flux reduction by up to 80% (cf. standard divertor). Edge transport modeling with UEDGE performed for NSTX Upgrade shows that the projected peak divertor heat fluxes 10-30 MW/m$^2$ can be mitigated with the SF divertor to tolerable levels [8]. Outstanding SF divertor questions include the null-region transport mechanism, its scaling with plasma collisionality, power, and $grad$ B, and radiative instability formation and thresholds. This work is supported in part by the US DOE under Contract DE-AC5207NA27344.

Recent experimental studies on the medium sized TCV tokamak focus on the physics of innovative divertor concepts including the snowflake divertor (SFD) and their utility for a fusion reactor. The longer connection length and larger divertor volume of a SFD in a reactor promise lower peak heat fluxes onto the divertor targets and, in particular, an easier access to detached divertor conditions. In a SFD, power losses can be distributed between three divertor legs, out of four, leading to a natural reduction of the total heat load at the targets. Moreover, it is predicted that the cross-field transport associated with the ExB particle drift velocities, in the null region, is enhanced compared to the singlenull divertor (SND) configuration leading to a further peak heat flux reduction. Measured target profiles support this mechanism. Compared to SND, experiments to increase the radiation fraction in the SFD in TCV have shown disparate results for increased deuterium fuelling and neon seeding, probably as a consequence of the TCV specific modifications of the scrape-off layer (SOL). An increase of the radiation with neon seeding in the SFD configuration may be the consequence of the increased SOL volume near the relatively hot null region that favors neon radiation, whereas the increase of the radiation with deuterium fueling, presumably by increasing the carbon content, in the SND might be the consequence of the increased volume of the relatively cold far-SOL which benefits to carbon impurity radiation. Any extrapolation of the observations in present devices to a reactor, however, has to be based on validated physics models. Progress in the validation process is being made by extending the suite of codes that can address the complex topology of the SFD and by enhancing the operational diagnostic capabilities of the TCV facility.
In the fusion research beyond ITER, a great effort is being carried out to study alternative solutions for the divertor configuration. The most challenging issue is related to the capability of the first wall materials to withstand the high thermal loads foreseen for DEMO. To address the problem, the experimental and modelling activities have followed two different approaches, to investigate: 1) alternative magnetic configurations aimed to reduce the heat loads by acting on the geometry of the field lines and 2) innovative materials for the target plates of the divertor such as liquid metals (Li, Sn) properly confined against MHD forces. In this framework, a two years program has been planned for testing on FTU a cooled liquid Li limiter (CLL) (2014&2015) and subsequently a cooled liquid Sn limiter (2016) with thermal loads as high as 10 MW/m² and up to 4.5 s of plasma duration. In 2015, these long pulses at Bₜ=4T and Iₚ=300KA were obtained for the first time in FTU. The CLL dedicated discharges were both ohmic and with auxiliary heating power (PₑCrH =500 kW). Circular and elongated shape (k ~1.2) were tested as well as different CLL positions under the TZM toroidal limiter shadow up to 1.8 cm inside the last closed magnetic surface in elongated plasmas. For almost all the conditions, the Li surface temperatures monitored by a fast infrared camera was maintained below the threshold of Li evaporation (~500 °C). Heat loads up to 2.3 MW/m² have been withstood by the limiter surface for all the duration of the plasma discharge (1.5s). However for the maximum insertion, some hot spots were observed localized on the joint points of the strips of CPS structure (maximum heat load of ~2MW/m²). Consistently non-negligible Li evaporation (500 - 570°C) was detected by monitoring the Li visible line intensity at 670.7 nm. Nevertheless no plasma disruption due to Li evaporation occurred and no damage of CLL surface was observed by a visual inspection “in situ” from the optical window of CLL volume. A first success has been attained by having recognized, as responsible for the observed hot spots, the misalignment of the W strips of the CPS structure that has been corrected by manufacturing a new active CLL refrigeration head in Red Star Labs. In this paper we will present FTU results obtained with the first CLL version and will comment on two important upgrades of the system that have been implemented in ENEA following the indications coming from the first experiments.
One of the main challenges in the European fusion roadmap is to design a heat and power exhaust system able to withstand the large loads expected in the divertor of a DEMO fusion power plant. Therefore, in parallel with the programme to optimise the operation with a conventional divertor based on detached conditions to be tested on the ITER device currently under construction in Cadarache, a specific project has been launched to investigate alternative power exhaust solutions for DEMO, aimed at the definition and the design of a Divertor Tokamak Test facility. This tokamak should be capable of hosting scaled experiments integrating most of the possible aspects of the DEMO power and particle exhaust. DTT should retain the possibility to test different divertor magnetic configurations, liquid metal divertor targets, and other possible solutions for the power exhaust problem. The DTT project proposal refers to a set of parameters selected so as to have edge conditions as close as possible to DEMO (in terms of the temperature and the normalized collisionality, pressure and ion gyro radius), while remaining compatible with DEMO bulk plasma performance in terms of dimensionless parameters and a given constraints. The talk will illustrate the DTT project proposal, referring to a 6 MA plasma with a major radius of 2.15 m, an aspect ratio of about 3, an elongation of 1.6-1.8, and a toroidal field of 6 T. This selection will guarantee to have a sufficient flexibility to test a wide set of divertor concepts and techniques to cope with large heat loads, including conventional tungsten divertors, liquid metal divertors, both conventional and advanced magnetic configurations (including single null, snow flake, quasi snow flake, X divertor, double null), internal coils for strike point sweeping and control of the width of the SOL (Scrape-Off Layer) in the divertor region, radiation control. The CS and PF coils are planned to provide a total flux swing of more than 35 Vs, compatible with a pulse length of more than 100 s. This pulse length, fully compatible with the mission of the study of the power exhaust problem is obtained using superconducting coils (NbTi for PF coils, Nb3Sn for the CS and the TF coils). Additional heating of 25 MW will be provided in the first phase of the operation using ICRH and ECRH. Afterwards, the ECRH heating power will be increased and NBI launchers will be added. The vacuum vessel is a single 35 mm shell of INCONEL 625, with five ports in each of the 18 sectors. The first wall is made of 5 mm of tungsten coating on a 60 mm stainless steel structure. The tungsten coating is thicker in selected zones of the first wall (where the plasma leans during the limiter phases of ramp-up and shut-down, where the plasma is expected to hit the wall in a disruption, at the upper strike points of double null configurations). Particular attention will be dedicated to the diagnostics and control issues, especially those relevant for plasma control in the divertor region, designed to be as compatible as possible with a DEMO-like environment. The construction is expected to last about seven years, and the selection of an Italian site would be compatible with a budget of 500 MEUR.
O-18: CONCEPTS FOR FAST FLOWING LIQUID LITHIUM WALLS AND DIVERTORS

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The use of liquid lithium plasma-facing surfaces in a tokamak fusion reactor may solve a wide array of issues with the use of high-Z solids, including erosion and dust formation, power handling and heat removal, confinement degradation and impurity accumulation, and tritium inventory control. The candidate concept discussed here is a free-surface, fast flowing liquid lithium wall system, with a flow velocity of 5-20 m/sec – sufficient to provide self-cooling (convective heat removal with the flowing liquid metal). The free-surface flowing lithium wall would be 0.5 to a few cm thick, and fully axisymmetric, and cover a significant fraction of the poloidal extent of the plasma. Lithium would flow over a guide wall. Since the guide wall itself is not exposed to plasma, and is cooled by the flowing lithium layer, it can be constructed of low thermal conductivity materials such as ferritic steel, which has shown good tolerance to fast fission neutron damage. J×B forces produced by externally driven poloidal currents in the liquid lithium, combined with the toroidal field, restrain the free-surface liquid lithium PFC against the guide walls. The flowing wall system described here incorporates an outflow region into a reservoir or catch basin, which also forms a lower single null divertor target. Liquid lithium would be recirculated within the toroidal field volume, using inductive (J×B) pumping in ducts, which replenish the free surface flow from the reservoir. Plasma heat is removed via a heat exchanger, operating at low power density, incorporated into the lithium reservoir. No other wall cooling system is envisioned, except for the need to locally cool RF launchers and other in-vessel service structures. Coolant from the heat exchangers would subsequently be routed through the breeding blanket, which can operate at significantly higher temperature than the flowing lithium PFC to provide good power conversion efficiency. As an initial step towards a full-wall fast flow system, we will describe a compact liquid lithium divertor concept which employs fast flow, J×B forces to restrain the liquid metal and drive flow, and recirculation within the toroidal field volume. The divertor concept is designed around a possible implementation, as a lower single null divertor target, in NSTX-U. The design goal would be a flowing lithium target capable of exhausting the full 14 MW NBI capability of NSTX-U.

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China Fusion Engineering Test Reactor (CFETR) is proposed as a good complement to ITER for demonstrating of fusion energy and now in conceptual design phase\(^ {1,2}\). As a key component in the superconducting Tokamak CFETR, divertor is directly faces the outflow particles and power from core plasma, therefore, heat exhaust and impurity screening abilities are desired as higher as possible. Although the expected fusion energy \( P_f \) of CFETR is 200 MW which is lower than ITER, due to a higher auxiliary heating power and smaller size, the heat flux flow into scrapped-off layer (SOL) is comparable with ITER. Further considering the smaller size of CFETR than ITER, the heat flux onto divertor targets may even exceed those of ITER.

To find a more effective way to exhaust the heat power, Snowflake\(^ {3}\) divertor is proposed recently and also now under consideration for CFETR along with the standard lower-single-null divertor. In our previous work\(^ {4}\), a density scan simulation by using SOLPS5.0 is performed on the quasi-snowflake divertor, and the peak heat load is found reduced even the flux expansion is not so broad. Because carbon is used as a substitute of seeded impurities such as Ar, in present work, Ar is injected to find a proper position and puffing rate which could achieve similar heat exhaust ability as carbon, while Ar impurity is well screened from the core plasma.

**Reference**

O20: ITER-like tungsten divertor development and experiments on EAST


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Plasma facing materials, especially for divertor target plates, have been recognized as a key issue in the magnetic fusion community. In order to withstand rapid increase in particle and power impact onto divertor and demonstrate the feasibility of the ITER design under long pulse operation, EAST upper divertor has been upgraded into ITER-like W/Cu mono-block structure with actively water-cooling system and steady-state heat removal capability of 10 MW/m² [1]. In the baffle and dome regions, flat type W/Cu plasma facing components (PFCs) also with actively cooling were utilized, which can withstand the heat load of 5 MW/M². The manufacturing W/Cu mono-block and flat type PFCs have been performed by the technological combinations of “HIP+HIP”, i.e., W armor joining with a pure copper interlayer by means of HIP technology and then the pure copper interlayer welding to CuCrZr heat sink by means of HIP technologies. To realize the seal joint of W/Cu mono-block target and baffle, the dome upper W/Cu/CuCrZr tile and lower CuCrZr heat sink, the cooling tubes and CuCrZr heat sink, the electron-beam welding (EBW) technique has been applied. In the 2014 EAST commissioning campaign, several leaks in the EBW “tube-plate” seams appeared during the baking of the device and the leaking of “plate-plate” seams occurred during plasma discharges, which have been technically optimized and repaired for the 2015 operation by controlling the seam quality of the W/Cu PFCs.

It was observed that W sources in EAST W divertor were significantly enhanced during NBI heating phase, intra type-I ELMs and disruption. Li aerosol injection in divertor can mitigate the W sputtering during NBI heating. The poloidal distribution of W source in W divertor corresponds to particle and heat fluxes obtained by divertor probes in some cases. With ion $\nabla B$ drift directing downwards, particle and heat fluxes favor the upper outboard divertor. Hot spots often appeared at tile edge along poloidal direction in W divertor, especially at the cassette module edge where the misalignment between neighboring tiles is larger and thus the W sputtering could be more serious.

In the ongoing 2015 campaign, the EAST W/Cu divertor has successfully withstand the baking up to 250 °C and the active cooling-water is circulating currently. The upper single null operation is scheduled late this campaign, and the forthcoming experimental results will also be presented.

References:
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Recent DIII-D experiments comparing the standard divertor (SD) and X-Divertor (XD) geometries show heat and particle flux reduction at the divertor target plate. The XD features large poloidal flux expansion, increased connection length, and poloidal field line flaring, quantified by the Divertor Index (DI = 1 for SD, and DI > 1 for XD). XD geometries were created at low triangularity, capable of utilizing the Divertor Thomson Scattering diagnostic, and at high triangularity on the lower floor with higher DI and leg length. Both SD and XD were pushed deep into detachment with increased gas puffing, until core energy confinement and pedestal pressure were substantially reduced. As expected, according to IR data, outboard target heat fluxes are significantly reduced in the XD compared to the SD under similar upstream plasma conditions, even at low Greenwald fraction. The high-triangularity XD cases show larger reduction in temperature, heat, and particle flux relative to the SD in all cases, while low-triangularity (shelf) XD cases show more modest reductions in temperature, heat, and particle flux over the SD. Consequently, heat flux reduction and divertor detachment may be achieved in the XD with less gas puffing and higher pedestal pressures. Further analysis, as well as detailed modeling with SOLPS, is underway to elucidate the underlying physics mechanisms. These initial experiments suggest the XD as a promising candidate to achieve divertor heat flux control compatible with robust H-mode operation in future tokamaks. This includes ITER, on which, despite stringent coil constraints, the design and modeling of an X-Divertor appears possible. The feasibility of an ITER XD will be discussed.

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Advanced magnetic configuration are characterized by the presence of two more or less spaced poloidal field nulls whose reciprocal position can give rise to a large family of different configuration with quite different role in the heat and particles exhaust. The overlap or the topological closeness of the two nulls can create a characteristic hexagonal (reminiscent of a snowflake – SF [1]) separatrix structure with a large region having B and its derivative close to zero. The magnetic field properties and the plasma behaviour in the snowflake are determined by the simultaneous action of both nulls, this generating a lot of interesting physics, as well as providing a chance for improving divertor performance. Another quite different configuration can be generated, with flaring flux lines in front of the divertor plates (XD configuration [2]). These two configurations can be seen as the two “extreme” cases of the so called quasi-Snow Flakes experimental scenarios [3]. In the recently proposed DTT facility, a set of small internal coils will have the capability to rule a given two null equilibrium (generated by the external coils), modifying the given global equilibrium from an XD up to a SF divertor configuration (Fig1).

Figure 1: Top left two nulls configuration obtained by the external coils; top center, the “hill” like field reference configuration has been varied to a monotone slop like field configuration (left bottom); top left, the “hill” like field reference configuration has been varied to a “mirrored field configuration (right bottom) including, of course, all the intermediate situations. In this paper we will illustrate this experimental actual possibilities, i.e. hardware possibilities and related scenarios. On a given set of possible divertor magnetic configurations we will illustrate the tolopagical properties and the possible physics advantages.

References
P-3: THE ERGODIC DIVERTOR CONCEPT
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In the late seventies the need for divertor control of plasma-wall interaction and the new development in the field of chaos led to the first ideas of using chaotic magnetic fields at the plasma boundary. In that respect, the Tore Supra ergodic divertor is the most elaborate concept for tokamaks that has been proposed. It included a set of in-vessel coils to induce a properly designed spectrum of the magnetic perturbation, actively cooled target plates, pumping capability and dedicated instrumentation for the experimental investigation of the diverted plasma properties.

As for standard divertors, the plasma facing components are radially withdrawn from the separatrix, which is the boundary surface between the closed field lines of the core and the open field lines in the edge. For a properly designed coil system, this only occurs for a given safety factor at the plasma boundary. Most experiments where performed at 3 T toroidal field with plasma current 1.4 MA, so that the region open field line region extended from the separatrix at q=2 to q=3 at the divertor coil. In that case the volume corresponding to the outer 20% of the maximum plasma minor radius, a_{max}=0.80 m, is dedicated to diverted field lines (35% of the plasma volume). Near poloidal and toroidal symmetry is achieved, which is a marked difference compared to the axisymmetric divertor.

Key experimental findings have been i) very strong screening efficiency driving plasma detachment at low core density, reduced impurity content, in particular for non-recycling impurities such as metallic impurities ii) low plasma temperature at the target plate (10 eV range) iii) unperturbed core properties compared to limiter plasmas with confined regions extending up to a_{max}. The large volume dedicated to diverted field lines proved efficient to achieve strong radiation in the divertor volume as well as important plasma pressure drop along the field lines.

The key and original property that emerged was the existence of three transport regimes along the chaotic field lines that we called laminar and ergodic transport regimes, as well as a self-healing region at the separatrix. The laminar regions extending from the actively cooled target plates are 3D Scrape-Off Layer (SOL) structures with radial penetration and a helical pattern around the plasma. The ergodic region extends from the separatrix to the radius of the laminar region. It connects with chaotic mixing the core plasma to the SOL laminar structure. Finally the self-healing region at the separatrix is such that in a narrow region, the plasma exhibits a sharp transition from the low temperature and density prevailing in the open field line region to values that would be reached in the limiter configuration at zero magnetic perturbation. Compared to the limiter L-mode operation, the ergodic divertor operation has demonstrated an efficient means to control plasma-wall interaction with no confinement degradation despite the large volume of open field lines.

Experimental evidence from JFT-2M, and recent experimental investigation with RMPs indicate that stochastic boundaries could require a larger H-mode threshold power. However, this regime is not ruled out.

The ergodic divertor can thus be regarded as an interesting concept that should be considered as alternative configuration for divertor studies.
ITER-like divertor will be made of thousands of tungsten mono-block tiles assembled in complex three dimensional geometry with shaping of individual tiles so as to intercept and extract the heat load partly arising due to magnetized plasma particles incident along scrape-off-layer at grazing angles to the divertor surface and partly arising due to non-magnetized energetic charge-exchange neutral particles as well as thermal radiation from SOL and core plasma region. As a result of this, fabrication and alignment of divertor targets for ITER-like tokamak is already considered to be a challenging task. Non-uniform heat flux on divertor target not only arises due to exponentially decaying power flux along SOL away from the separatrix, but also due to finite fabrication tolerances and misalignment of divertor targets in poloidal and toroidal directions. In addition to non-uniform heat flux, transient events in plasma and SOL results in transient heat flux on divertor targets. Therefore, it has become essential to study performance of divertor targets under non-uniform and transient thermal load conditions.

Present paper describes recent experimental investigations performed to study thermal behavior of tungsten mono-block divertor targets under non-uniform and transient thermal load conditions. Engineering analysis and computations performed to study thermo-mechanical stress on divertor targets are also presented in the paper.
The current design of a divertor for a fusion device like ITER or a DEMO expects heat loads in the order of 5 – 20 MW/m² in stationary operation. Thermal shocks induced by off-normal events (disruptions) or during normal operation (edge localized modes, ELMs) are superimposed on the stationary load impinging on the plasma facing material (PFM). In total this leads to a surface temperature determined by the stationary heat load (SHL) at the respective location and sudden temperature excursions on top induced by the transient heat loads (THL). The spatial and temporal temperature variations cause changes of the stress fields in the PFM. These can lead to immediate or fatigue-induced morphological changes (roughening, cracking, melting) depending on the intensity and frequency of the THLs. The impact of particles has additional effects like blistering or growth of nanostructures which can in turn influence the thermal shock response. All of these PFM changes can influence the performance and lifetime of the plasma facing component (power handling capability, loss of coolant or PFM) or of the reactor (plasma contamination by dust or melt splashing).

This paper reports about the current activities at Forschungszentrum Jülich on the simulation of such transient events using electron beam and laser devices, also combining them with stationary particle exposure. When comparing electron beam and laser experiments, the larger penetration depth of the electron beam (≤ 7 μm for ≤ 120 keV in W, in the order of nanometres for the laser) did not have an influence on the material response. Experiments using a hydrogen plasma background led to blistering, which could be reduced by simultaneous thermal shocks. Helium plasma was used to create nanostructures (“fuzz”), which increased the laser absorption and hence influenced the thermal shock response. Finally, the investigation of various tungsten grades under THL showed that the threshold for any morphological change can be very low: As low as 0.19 GW/m² (∆t = 1 ms) for 100 thermal shocks or lower for a higher number of thermal shocks. This means, either the plasma control has to be improved to a level that guarantees an operation with only few weak thermal shocks or new technologies have to be developed to repair divertor components without removal from the machine in order to keep operational availability high. For this purpose, first proof-of-principle experiments are shown which demonstrate that it might be possible to repair thermal shock induced damages by laser treatment.
Operational conditions of divertor target in next step fusion devices are more severe in comparison with ITER. Current divertor designs and technologies have a limited application to these conditions and so new design concepts/technologies are required. The main reasons which practically prevent to use traditional motionless solid divertor target are analyzed. Several alternative divertor target concepts are described. Comparative analysis of these concepts (advantages/drawbacks) is made and prospects for their practical implementation are prioritized.

The concept of swept divertor target with liquid metal interlayer between moving armour and motionless heat-sink is presented in more details. The experimental plan to solve critical issues of this design is presented.
A new divertor concept consisting of Vertical Free-Surface Streams (VFSS) of liquid metal as shown in Fig. 1, where ball chains are used to form the VFSS, is proposed. Arranging the arrays of liquid metal VFSS, a Vertical Louver Divertor (VLD), as shown in Fig. 2, can be formed (Japanese patent application No. 2015-112782). The divertor plasma flowing along the magnetic field line slanted through the VLD cannot penetrate the VFSS. However, once the plasma becomes recombined on the liquid metal surface, the neutrals can be pumped out to the back of the VLD (see Fig. 2(b)). At this moment, pure tin (Sn) is considered as the first candidate of the liquid metal. The low melting point of ~500 K, the low vapor pressure (1 Pa at ~1500 K), and the low material cost are the merits of tin. Other candidates, for example, an alloy of tin-lithium and molten salt, are also worth considering. As for the pumping, a new device named the Supersonic Jet Pump (SJP) (Japanese patent application No. 2015-112781) is being developed. The SJP is, in essence, a cryopump without the regeneration process. Hydrogen isotopes become condensed on the surface of a cooled central shaft. Then, a cylindrical cutter shaves off the solid hydrogen isotopes and pushes them out of the pump. The tritium inventory in the SJP can be minimized. A part of the shaved ice is sublimated inside the pump and puffed to the central shaft through supersonic nozzles. Hydrogen clusters formed in the supersonic flow will give its kinetic momentum to the remaining neutrals. This “diffusion pump effect” is expected to be effective for helium pumping.
P-8: How to obtain ductile tungsten

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Tungsten and tungsten-based materials are main candidates for divertors as these materials show outstanding high-temperature properties such as a high melting point, good thermal conductivity, high temperature strength and a high threshold against sputtering. However, some mechanical properties such as fracture toughness and ductility make the application of tungsten quite challenging for fusion environments. Former approaches to improve the mechanical properties, such as alloying with rhenium, are not pursued as intense anymore. No easily available alloying element as a replacement for rare and expensive rhenium has yet been found.

Turning away from alloyed tungsten, the adjustment and stabilization of a beneficial microstructure of – more or less – pure tungsten is a very promising method. An increased number of steps of warm- and cold-rolling during manufacturing of half-finished tungsten products lead to wires and foils. These deformation steps change the microstructure in a way so that mechanical properties are improved, at least for two out of three testing directions. In the case of thin tungsten foil, the microstructure consists of thin, pancake-shaped grains; in case of wires the material contains needle-shaped grains.

This contribution focuses on the increased fracture toughness of thin tungsten foils, which were tested at low temperatures and at fusion-relevant temperatures. The outcome of these experiments emphasizes the importance of the manufacturing history (i.e. microstructure) on the mechanical properties. Most importantly, these results are directly related to the correct design of tungsten components, taking into account strong and weak directions of tungsten based materials.
P-9: ON THE FEASIBILITY OF A MAGNETIC NOZZLE AS THE MATERIAL-LESS DIVERTOR FOR TOKAMAK FUSION REACTORS

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Containment and mitigation of the intense heat fluxes from ITER and reactor-like tokamaks is a well-established and unresolved issue for steady-state magnetic fusion. The thermal plasma heat flux densities in the scrape-off layer (SOL) are about 30 MW/m² in the existing machines. The heat flux density in ITER-FEAT is expected to be about 10 times higher. The heat flux in a fusion reactor, which has a Q above 20, exceeds that of ITER-FEAT, which only has a Q of 10. These heat flux densities easily exceed the tolerance of any known material even with active cooling. We investigate the feasibility of a material-less divertor for magnetic fusion, specifically a magnetic nozzle. A supersonic, converging-diverging magnetic nozzle can convert the SOL thermal energy to exhaust kinetic energy which in turn can be utilized by a magnetohydrodynamic (MHD) generator to produce useful power. Furthermore, such concept isolates the main plasma region from the divertor region reducing impurity back flow from plasma-wall interactions. Based on relatively straightforward geometry and magnetic field strength variations the mass flow rate and heat dispersal can be adjusted. Lastly, adequate pumping of the exhaust can be achieved by using proper material geometry downstream. In order to conceptually quantify the benefits of a material-less divertor a preliminary magnetic nozzle design is presented accounting for variable nozzle efficiency and subject to two requirements that control the plasma exhaust temperature and address possible structural limitations. Furthermore, we present some basic calculations that quantify potential electric power generation via a MHD generator also subject to variable efficiency. In particular, the model utilizes typical SOL stagnation plasma conditions of 3.5 atm and 100 eV. Based on reasonably assumed nozzle and MHD conversion efficiencies the analysis predicts power generation on the order of 0.3 GW to 1.5 GW with comparable power dissipated during plasma expansion. Based on some basic circular electromagnet configuration in toroidal geometry these dissipated power levels translate to heat flux densities approximately 100-500 MW/m² with minimum plasma exhaust temperature values as low as 4 eV. Additionally, and based on a condition imposed in the model the MHD power generated is adequate to power the electromagnets in a scenario of a self-sustained magnetic nozzle.
Detached divertor-plasma operation, where a large fraction of the core exhaust power is radiated before striking the target plates, is attractive for limiting the peak plate heat flux. Such plasmas have electron temperature ~1 eV near the plate. Changing the position of the separatrix strike points on the geometrically varied DIII-D target plates is allowing a systematic study of how plate shape impacts accessibility to detached operation. Reported here are 2D plasma/neutral transport simulations of these configurations using the UEDGE code including cross-field drifts and impurities. Results are reported on how detachment onset scales with the angle between the magnetic flux surface and the divertor plate, wall pumping of neutrals, separatrix density, and core power. A key parameter to aid in the prediction of detachment onset is the total particle inventory in the scrape-off layer. Different initial conditions are found to sometimes yield different steady-state solutions for identical input parameters, one being an attached plasma and the other detached. The temporal evolution from attached to detached plasmas as the separatrix density is increased is illustrated. Comparisons are made between simulation results and published experimental measurements.
Horizon 2020 is the largest EU Research and Innovation programme to date. The European fusion research programme for Horizon 2020 is outlined in the “Roadmap to the realisation of fusion energy” and published in 2012. As part of it, the European Fusion Consortium (EUROfusion) has been established and will be responsible for implementing this roadmap through its members. The European fusion roadmap sets out a strategy for a collaboration to achieve the goal of generating fusion electricity by 2050. It is based on a goal-oriented approach with eight different missions including the development of heat-exhaust systems which must be capable of withstanding the large heat and particle fluxes of a Fusion Power Plant (FPP). The main aims of the Mission for a solution on heat-exhaust systems and current up to date progress will be summarised. This includes a technological study of feasibility and performance of water-cooled divertor targets concepts, which extend the ITER design and technology to DEMO relevant condition (e.g., higher coolant temperatures, pressures and higher n-dose) as well as assessment of the adequacy for DEMO and proof-of-principle tests of innovative divertor concepts and geometries (such as super-X and snowflake configurations and the use liquid metals based divertors). Finally, as part of the coherent mission approach, the definition of the exact scope and technical specifications of a Divertor Tokamak Test (DTT) facility (either a new facility or the upgrade of existing facilities) will have to be completed and, after a thorough review, a decision should be taken in 2016 for its construction. The EUROfusion consortium strategy to set up an efficient Work Breakdown Structure and the collaborative efforts to address these challenges will be also presented.
This work is included in an ongoing co-ordinated effort in EU to optimize the DEMO divertor design aiming to satisfy plasma physics, materials and pumping technology needs in an integrated approach. In a first principle step, two extreme limit cases of the DEMO ITER-like divertor design with and without dome are analyzed in terms of pumping efficiency by means of the DIVGAS code [1], which is based on the Direct Simulation Monte Carlo method [2]. The gas flow pattern inside the private flux region is illustrated and macroscopic parameters (pressure, temperature, density) as well as the overall conductance of the sub-divertor structure are provided. The imposed boundary conditions consist of the given pressure and temperature at the interfaces with the plasma fan and of the prescribed sticking coefficient at the defined pumping surfaces [3]. The density (i.e. pressure) dependence of the effective pumping speed has been taken into account. Calculations show that the required throughput can be achieved in the DEMO divertor configuration with and without dome but is necessitating to establish different pumping conditions (number or location of required pumps). The effect of the dome on density compression is more pronounced in the medium density regime whereas at higher densities and dominant neutral-neutral collisions the difference is moderate.

References
P-13: Study of the influence of neutral-neutral collision on CFETR Divertor by using SOLPS-ITER

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Lower-single-null Divertor, which is the baseline in the conceptual design of China Fusion Engineering Test Reactor (CFETR) [¹,²], has been studied by using SOLPS5.0 [³]. By using carbon as a substitute of seeded impurity, along with the increasing upstream density (increasing D₂ gas p puffing rate in simulation), a change of the divertor operational status from low recycling regime to high recycling regime and finally detached regime is clearly shown in the simulation results while the peak heat loads onto both inner and outer divertor could be reduced below 10 MW/m² with a good screening of the carbon impurity. However, due the high density in the divertor, neutral-neutral collision should not be neglected.

A simulation for ITER by using SOLPS4.3 clearly shows the influence on the density and temperature near the divertor target and thus the peak heat load [⁴]. The latest SOLPS version SOLPS-ITER [⁵] is released recently, where the MPI parallelized EIRENE code is coupled. In this work, by using SOLPS-ITER, the CFETR lower-single-null divertor is simulated to study the influence of neutral-neutral collision. Based on the simulation results, more reasonable operational window is given for the lower-single-null divertor in CFETR.

Reference