Third IAEA Technical Meeting on Divertor Concepts

Monday, 4 November 2019 - Thursday, 7 November 2019

IAEA Headquarters, Vienna, Austria

Programme

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Monday 04 November 2019

Registration - Board Room C (C Building, 4th Floor) (4 Nov 2019, 08:30-09:00)

Arrival of participants, distribution of badges, possibility to register and pay for events.

Welcome and Opening - Board Room C (C Building, 4th Floor) (4 Nov 2019, 09:00-09:30)

Welcome addresses by the IAEA and by Programme Committee.

-Conveners: Denecke, Melissa (IAEA); Barbarino, Matteo (International Atomic Energy Agency); Leonard, Anthony W. (USA)

Plasma Facing Component Design - Board Room C (C Building, 4th Floor) (4 Nov 2019, 09:30-10:20)

Oral sessions collect all contributions invited or accepted by the Programme Committee for a complete in depth plenary session. The reserved time slot for 30' talks is 25' for presentation and 5' for discussion. The reserved time slot for 20' talks is 17' for presentation and 3' for discussion.

time	[id] title
09:30	**[18] Development and testing results of water-cooled divertor target concepts for EU DEMO reactor** *Presenter: VISCA, Eliseo (ENEA)* Power handling is one of the most critical scientific and technological challenges for a nuclear fusion power plant. Divertor is a key in-vessel component of a fusion reactor being in charge of power exhaust and removal of impurity particles. For the European demonstration reactor (EU-DEMO), divertor targets local peak heat flux is expected to reach more than 20 MW/m2 during slow transient events. Sufficient heat removal capacity of divertor targets against normal and transient operational scenarios is the major requirement. Material degradation due to neutron irradiation has to be also considered (cumulative dose for the structural material: up to 14 dpa for lifetime). To find a feasible technological solution, an integrated R&D program for the European DEMO reactor was launched since 2014 in the framework of the EUROfusion Consortium Work package “Divertor” (WPDIV). The preconceptual phase was concluded in 2018 where six different water-cooled target concepts were developed and evaluated. For all concepts a common R&D approach was adopted, namely, design study, failure simulation, design rules, materials definition, mock-up manufacturing, non-destructive inspection, high-heat-flux tests and microscopic examination of damage. Textured pure tungsten was used as reference armor material. Each concept was characterized by 1) tungsten wire-reinforced copper composite heat sink, 2) thin and thick graded interlayer, 3) thermal barrier interlayer, 4) flat-tile armor with W-Cu composite heat sink block, and 5) conventional ITER like monoblock, respectively. Extensive high-heat-flux tests were performed under DEMO-relevant operational condition (heat flux: 20MW/m², coolant: 130°C, 4MPa) up to 500 load cycles after screening test up to 25MW/m². Furthermore, overload tests were carried out to explore the maximum loading limit (heat flux: >25MW/m², coolant: 20°C, 4MPa). The final results of R&D program are presented focusing on the overall technology achievement in this preconceptual phase highlighting materials technology, mock-up fabrication and high-heat-flux qualification accompanied by non-destructive inspection and microscopic damage examination. The comparison of results of each design concepts candidate is presented. Correlation of non-destructive test and eventual thermal performance degradation is reported with evidences coming from metallographic investigations.
10:00	**[23] New developments in the design of a helium-cooled divertor for the Europen DEMO** *Presenter: GHIDERSA, Bradut-Eugen (Karlsruhe Institute of Technology)* The recent development of novel refractory materials for fusion applications with better ductile properties has led, in the recent years, to rethink the helium cooled divertor design under development at KIT. In particular, the availability of pipes and plates made of tungsten laminates showing very good mechanical properties over an extended temperature range has triggered a new search for concepts in which such materials can be used. As a first attempt, a divertor concept having a similar geometry with the water-cooled divertor developed for ITER was investigated both numerically and experimentally. The concept uses W slabs as armor brazed on a 15mm in diameter W-Cu laminate tube. The heat deposited on the surface of the armor is removed using a jet-impingement cooling scheme. Thus, the jets are created using a coolant distribution manifold in the form of a 6mm in diameter cartridge installed inside the W-laminate pipe. Under this configuration, the tested mock-up was able to withstand 8MW/m² for a total duration of 83h (100 cycles, each 5 min long) and 10MW/m² for more than 2h (25 cycles, each 5 min long). Taking advantage of the lessons learned with the first mock-up, a second mock-up has been developed using an improved cooling scheme, both in terms of pressure losses and in terms of jet flow distribution. The paper will give an overview on the experimental results obtained so far and discuss the further steps foreseen in the development of the present concept, including the integration into a divertor target.

Coffee Break - Board Room C (C Building - 4th Floor) (10:20-10:40)

Plasma Facing Component Design: Continued - Board Room C (C Building, 4th Floor) (4 Nov 2019, 10:40-11:20)

Oral sessions collect all contributions invited or accepted by the Programme Committee for a complete in depth plenary session. The reserved time slot for 30' talks is 25' for presentation and 5' for discussion. The reserved time slot for 20' talks is 17' for presentation and 3' for discussion.

time	[id] title
10:40	**[75] Additive manufacturing of tungsten by means of laser powder bed fusion for plasma-facing component applications** *Presenter: VON MÜLLER, Alexander (Max-Planck-Institut für Plasmaphysik)* Tungsten (W) is currently considered the preferred plasma-facing material (PFM) for future magnetic confinement thermonuclear fusion reactors. This is mainly due to the fact that W exhibits a high threshold energy for sputtering by hydrogen isotopes as well as a low retention of tritium within the material. From an engineering point of view, however, W is a difficult metal to work with, as it is intrinsically hard and brittle which makes the processing and machining of W laborious and expensive. This is also the reason why rather simple geometries, for example flat tiles or monoblocks, are typically used for W armour parts in plasma-facing components (PFCs). Against these limitations, additive manufacturing (AM) technologies could represent a versatile and up-to-date approach for the realisation of W parts for PFC applications. The characteristic feature of AM processes is that three-dimensional objects are created by sequential layerwise deposition of material under computer control which means that such a technology is well-suited for producing objects with complex geometry. In this context, the present contribution will summarise results of research work conducted by the authors during recent years regarding AM of pure W by means of laser powder bed fusion (LPBF). In more detail, results will be presented regarding parametric material manufacturing investigations. These investigations include parameter variations regarding laser exposure and substrate preheating temperature as well as experiments performed on different LPBF facilities and with differing raw powder materials. In general, the investigations showed that bulk pure W can be consolidated directly by means of LPBF with a comparably high relative mass density of approximately 98%. However, it was also found during these investigations that W consolidated by means of LPBF can exhibit defects that are frequently encountered with respect to LPBF processing, such as porosity or crack formation due to the fact that LPBF processes typically induce high thermal gradients during material consolidation. The suppression of such defects is still considered a challenging issue with regard to LPBF of high melting point refractory metals like W. Furthermore, the contribution will present the latest results regarding AM of more complex and thin-walled W structures. In this context, two possible PFC applications will be discussed. On the one hand, the fabrication of tailored preforms for the manufacturing of tungsten-copper (W-Cu) composite material structures will be described. If such structures with optimised W-Cu material distribution are applied at the heat sink level of PFCs they can enhance the performance and integrity of such components notably. On the other hand, the fabrication of tailored anisotropic W lattice structures will be discussed. Such structures are currently of interest regarding DEMO first wall high-heat-flux limiter PFCs.

11:00 **[36] Damages on tungsten plasma facing components after experimental campaigns in WEST**

Presenter: FIRDAOUSS, Mehdi (CEA/IRFM)

Interaction between the tungsten, used as plasma facing material, and the D/He plasma is still under investigation, in particular for the long duration pulses (> 30s) and high heat fluxes (>10MW/m²). One of the main goal of WEST experiment is to study the behavior of an ITER-like divertor in a tokamak environment, and in particular its impact on the operation, for this type of conditions.

During the first phase of WEST operations (2017-2019), the divertor is made of inertial graphite tiles with a thin W coating, except for a dozen of components, all localized on a particular sector. These components, so called Plasma-Facing Unit (PFU) are made of W monoblocks assembled on a CuCrZr tube, and cooled by water. Seven different industrials manufactured the present 14 PFU, which exhibit different options, like sharp or chamfered edge. During the next phase, planned for 2020, the entire divertor will be equipped with 456 PFU, in order to be able to reach the power and duration expectations.

At the end of the last campaign (C3 – early 2019), these PFU have been closely observed, both inside and outside the tokamak. Several type of changes in the plasma-facing surface have been noticed, with different levels of damages. The most visible evolution is the general plasma footprint, due to the erosion / deposition mechanism. In WEST, its pattern is easily noticeable, due to the quite important ripple in the magnetic field.

However, more important damages have also been observed, like crack network and even local melting. Those damages are related to the relative vertical misalignment of the PFU, which has been precisely measured using a robotic arm and reaching 0.8mm. Triangular marks corresponding to optical hot spot (particles passing through the toroidal gaps), with several millimeters long cracks, have also been observed on several locations. Finally, the most impressive damage observed is a several centimeters long melted edge, on a location far from the strike points. This particular melting is attributed to a singular event, like disruption or run-away electron, which occurred near this trailing edge.

In conclusion, it is very noticeable that such damages were observed on the W PFU for the relatively low incident heat flux estimated (less than 3MW/m²). This is largely due to the important misalignment of the PFU. Even if those damages did not impede the last experimental campaign, PFU have been aligned with a tolerance in line with ITER specifications (< 0.3mm) for the next campaign, which will be more struggling in terms of deposited power on the divertor. This will allow to assess the impact of damaged components on advanced plasma mode, i.e. H-mode with ELMs.

Discussion Session: PFC Components - Board Room C (C Building, 4th Floor) (4 Nov 2019, 11:20-12:10)

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

Divertor Plasma Control - Board Room C (C Building, 4th Floor) (4 Nov 2019, 12:10-13:10)

time [id] title

12:10 **[59] Using variations in divertor magnetic topology and geometry to optimize divertor detachment**

Presenter: LIPSCHULTZ, Bruce (University of York)

Total flux expansion, a divertor magnetic topology design choice embodied in the Super-X divertor, is predicted through simple analytic models [1] and SOLPS calculations [3] to reduce the plasma and impurity density detachment thresholds as the outer divertor target strike point position, Rt, is increased. Since the total magnetic field, |B| ~ 1/R, |B| at the target is lowered as Rt is increased. Those predictions are contradicted by recent TCV experimental results [2]. The SOLPS-ITER code was utilized to both match TCV results and determine, more generally, the relative effect of detachment threshold by 'magnetic topology' (in this case, total flux expansion) and 'divertor geometry', for which we vary a) the separatrix incidence angle to the divertor surface and b) the effect of physical baffles that retain more neutrals in the divertor.

We quantify the role of those neutral effects through applying a definition of neutral trapping in form of the fraction of the total target ion flux that is re-ionized in a divertor flux tube which is just outside the separatrix - eta_RI. The analysis for TCV showed that neutral trapping was reduced as Rt was increased, negating the effect of total flux expansion on the detachment threshold.

This study has pointed a path towards more quantitative divertor design tools. Quantitative prediction of the effect of total flux expansion on the detachment threshold [1] and target temperature [3-4] already exist. The relationship between divertor geometry and the detachment threshold has been more qualitative: The angle between the separatrix and the target (effect a) is well known to affect the detachment threshold [5,6]. Baffling the divertor (effect b) is oft-used to protect the core from neutral effects as opposed to maximize divertor ionization. The quantitative relation between eta_RI and the detachment threshold for the two divertor geometry design choices will be shown and contrasted to that of total flux expansion. We will also discuss how the same design tools might affect other divertor design criteria - e.g. detachment control.

[1] B. Lipschultz et al, Nucl. Fusion 56 (2016) 056007
[2] C. Theiler, B. Lipschultz , et al., Nucl. Fusion 57 (2017) 07200
[3] D Moulton, J Harrison, B Lipschultz & D Coster, Plasma Phys. Control. Fusion 59 (2017) 065011
[4] T. Petrie et al., Nucl. Fusion 53 (2013) 113024
[5] B. Lipschultz et al., Fusion Science Tech., 51, (2007) 369
[6] M. Groth et al, J. Nucl. Materials 463 (2015) 471

12:30 **[68] The Small Angle Slot Divertor Concept for Steady-State Fusion**

Presenter: GUO, Houyang (General Atomics)

A major challenge facing the design and operation of future high-power steady-state fusion devices is to develop a robust boundary solution with an order-of-magnitude increase in power handling capability relative to present experience, while having acceptable erosion at the surface of the plasma facing components (PFCs) to ensure an adequate reactor lifetime. Recently, a small angle slot (SAS) divertor concept has been developed to enhance neutral cooling across the divertor target by coupling a closed slot structure with appropriate target shaping in the near SOL [1]. First results from DIII-D [2] find that cross-field drifts can have a significant effect on particle fluxes and pressure enhancement in the SAS, favouring operation with the ion grad-B drift away from the X-point, as currently employed for advanced tokamaks. SAS allows for transition to low temperature moderately detached divertor conditions with Te □ 10 eV at very low main plasma densities, lower than are usually attainable at all in the DIII-D high confinement (H-mode) plasmas used in these tests. In addition, pedestal performance and core confinement are significantly improved, and the final confinement collapse associated with the onset of an X-point MARFE occurs at significantly higher pedestal densities than for other divertors in DIII-D, thus widening the window of H-mode operation compatible with a dissipative divertor. For operation with the ion grad-B drift toward the X-point, the divertor plasma transitions to a bifurcative detached state at much higher densities, similar to other divertor configurations in DIII-D. Initial modelling with SOLPS-ITER including drifts shows that SAS can achieve a (highly dissipative) detached state across the target for the ion grad-B drift away from the X-point, while remaining (partially) attached for the opposite drift direction at a given upstream separatrix density, qualitatively consistent with the experimental observations. The modelling highlights how plasma drifts interact with closed divertor structures affecting overall particle transport and divertor dissipation. SOLPS modelling for CFETR [3] has shown that a SAS-like divertor configuration can also improve divertor dissipation, achieving Te < 5 eV across the divertor target plate, which is essential for steady-state operation, with respect to an ITER-like divertor configuration where Te remains high away from the strike point. Efforts are being made to develop the modeling tools needed for rapid evaluation of 2D-fluid divertor simulations and to develop reduced models to gain physics insights for use in guiding the divertor design and optimization for CFETR, as well as the next-step steady-state high-power density fusion device under consideration in the US.

This material is based upon work supported by the Department of Energy under Award Number DE-FC02-04ER54698.

[1] Guo H.Y. et al 2017 Nucl. Fusion 57 044001
[2] Guo H.Y. et al 2019, First experimental tests of a new small angle slot divertor on DIII-D, Nucl. Fusion https://doi.org/10.1088/1741-4326/ab26ee
[3] Ding R. et al, “Recent progress on divertor physics design of CFETR”, this workshop.

12:50 **[42] Latest advances in active control of H-mode detachment and its physics on EAST for ITER/CFETR**

Presenter: WANG, Liang (Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP))

Active handling of excessively high heat load and tungsten sputtering on divertor targets is of critical challenge for EAST and future fusion devices like ITER and CFETR. It is acknowledged by the fusion community that divertor detachment is the most promising means for steady state plasma-wall interaction control.

Significant progresses on the active feedback control of H-mode detachment for long-pulse high-performance operations have been made in EAST with ITER-like tungsten divertor in the last two years. A serial of methods and systematic experiments for detachment control including radiation power, particle flux roll-over, Te < 5 eV have been successfully developed, respectively. Both neon and argon impurity seeding have been demonstrated in maintaining the divertor detachment stably. More importantly, it has been observed that the core plasma performance exhibits no significant loss or even a slight increase during the stable feedback control phase, suggesting excellent divertor-edge-core plasma integration. A new periodic detached-attached dithering regime, which is of potential application for steady state operations, was clearly identified and reproduced with different target-Te values. A new detachment feedback control scheme, which combines divertor Te and radiation signals into feedback to achieve sustained detachment, has also been demonstrated successfully. It is indicated that the ExB drifts and other SOL flows play key important roles to the success of the detachment access and long pulse maintenance. The detachment density threshold exhibits a strong dependence on the divertor closure. The radiation feedback control for detachment was demonstrated in DIII-D high beta_p H-mode scenario within the joint DIII-D/EAST Task Force. Further experiments to demonstrate the latest EAST results of drift effect on detachment is scheduled in September on DIII-D in the high beta_p scenario with different divertor geometry. New advances that may arise will also be presented. These latest detachment advances and related physics understanding are of great importance to the design of CFETR divertor and the steady state operation of ITER.

This work was supported by the National Key Research and Development Program of China (No. 2017YFE0301300), the US Department of Energy (No. DE-FC02-04ER546981), the Anhui Natural Science Foundation (No. 1808085J07).

[1] K. Wu, Q. P. Quan*, B. J. Xiao, L. Wang* et al., Nucl. Fusion 58, 056019 (2018)
[2] L. Wang* et al., Nucl. Fusion 59, 086036 (2019)
[3] J. B. Liu, L. Wang* et al., submitted to Nucl. Fusion (in revision)

Lunch - Board Room C (C Building - 4th Floor) (13:10-14:50)

Divertor Plasma Control: Continued - Board Room C (C Building, 4th Floor) (4 Nov 2019, 14:50-15:10)

time	[id] title
14:50	**[58] X-point radiation and detachment control at ASDEX Upgrade**

Presenter: BERNERT, Matthias (IPP Garching, Germany)

Future fusion reactors require a safe, steady state divertor operation. In the detached regime, the power and particle fluxes to the divertor targets are sufficiently reduced to meet the material limits. In H-mode operation at the full-tungsten ASDEX Upgrade tokamak (AUG), this is achieved by injection of significant amounts of nitrogen into the divertor volume. This increases dominantly the divertor radiation and radiated power fractions of up to 90% were achieved at high heating powers (Pheat/R=12-13 MW/m).

In these conditions, the divertor is fully detached and the dominant radiation is emitted from a small, poloidally localized volume in the vicinity of the X-point. This X-point radiation is observed to vary its location relative to the X-point depending on seeding and power levels. Unlike the typical observation of such radiation at the X-point, identified as a MARFE, the scenario is stable and persisted against variation in heating power or seeding. The radiator is observed to move up to 15 cm inside the confined region, which corresponds to a normalized poloidal flux of ϱpol≈0.985. A further movement finally leads to a disruption of the plasma. The accessibility and implications of such a regime will be discussed in the view of future devices like ITER and DEMO.

The stability of the X-point radiator and the dependence of its position to external parameters such as N seeding level and heating power allow an active control of this scenario. This will allow the real-time control of a fully detached divertor, where target measurements are not sufficient any more. The sensor of the control is based on an array of horizontal channels of AXUV diodes, detecting the vertical position of the radiator. Similar to the existing controller on the divertor temperature [1], the nitrogen seeding level is used as actuator. The performance of this controller will be presented and compared to the existing controller for the divertor temperature at AUG and control schemes applied at other devices, such as DIII-D [2] and TCV [3].

[1] A Kallenbach et al 2010 Plasma Phys. Control. Fusion 52 055002
[2] D. Eldon et al 2017 Nucl. Fusion 57 066039
[3] T. Ravensbergen et al 2019 Submitted to Nuclear Fusion

Discussion Session: Divertor Control - **Board Room C (C Building, 4th Floor) (4 Nov 2019, 15:10-16:10)**

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

Coffee Break - Board Room C (C Building - 4th Floor) (16:10-16:30)

Poster Session I - Board Room C (C Building - 4th Floor) (4 Nov 2019, 16:30-18:30)

All submissions accepted as "Poster" should provide a poster conforming to the rules published in the meeting announcement. Posters will be shown outside of Board Room C near by the coffee area.
Board numbers correspond to Indico [ID] numbers.

[id] title	board
[44] Advanced divertor detachment in H-mode and baffled TCV plasmas *Presenter: THEILER, Christian (EPFL-SPC)* Operating a tokamak fusion reactor in a highly dissipative detached divertor regime, whilst maintaining sufficient core confinement, remains a major challenge. It is likely that advanced divertor solutions will be required. Proof of principle experiments of such ideas, as well as the underlying physics processes, are being explored on the TCV tokamak. Previous studies in L-mode revealed a lower detachment threshold with increasing divertor leg length, ascribed mainly to changes in the SOL width, and deeper detachment and improved control of the CIII radiation front with increasing poloidal flux expansion (fx). In the Snowflake minus, as predicted by simulations, the position of the X-point radiator could be displaced outside of the core plasma, although benefits in terms of core confinement could not be demonstrated. Increasing the outer target major radius (Rt) by 70%, seemingly the most direct way towards improved detachment characteristics, did not reveal the expected benefits. These studies have now been extended to neutral beam heated, ELMy H-mode [1,2]. For a fixed core shape and X-point position, the L-H power threshold was found to be weakly dependent on divertor geometry. Partial detachment of the outer target could be achieved, although only at levels of nitrogen seeding that are marginally compatible with H-mode confinement. More resilience of the H-mode regime to seeding and a stronger reduction in target particle flux was observed when increasing fx by a factor 4, while a 40% increase in Rt revealed approximately 20% slower upstream movement of the CIII front. The overall dependences on divertor geometry are, however, found to be relatively modest in these plasmas. These studies are expected to strongly benefit from the ongoing installation of divertor baffling structures on TCV. These baffles, which are compatible with a large variety of advanced divertor geometries [3,4], are expected to increase fuel and impurity compression in the divertor by an order of magnitude [4,5], which should facilitate detached H-mode operation. Furthermore, the baffles will aid in equalizing neutral trapping between different divertor magnetic geometries, found in recent SOLPS-ITER simulations to be necessary to isolate the effect of magnetic geometry, and in particular to recover the benefits expected of increasing Rt [6]. First results in baffled, advanced divertor geometries and comparisons to these predictions will be presented. [1] J. Harrison et al., PPCF 61, 065024 (2019) [2] C. Theiler et al., 27th IAEA-FEC, Gandhinagar, India (2018) [3] H. Reimerdes et al., Nucl. Mat. Energy 12, 1106-1111 (2017) [4] A. Fasoli et al., NF (2019), submitted [5] M. Wensing et al., PPCF, https://doi.org/10.1088/1361-6587/ab2b1f, (2019) [6] A. Fil et al., PPCF, ready to submit	

[29] Flat Tungsten High Heat Flux Components Development Based On Different Technologies

Presenter: YAO, Damao (Institute of Plasma Physics Chinese Academy of Sciences)

Divertor with tungsten act as plasma facing material is expected apply for future fusion reactors. How to exhaust high heat load deposit on tungsten is key issue. Tungsten monoblock structure which applied for ITER divertor is a very good solution. Is there any other tungsten bonding structure which can handling 10MW/m2 heat load or even more?
A kind of flat heat sink with special cooling channels structure, and tungsten slices bonded to it by different technologies is under developing. Analyses shows different tungsten thickness with different heat flux heat deposit surface temperature as table 1
Table 1. Analyses result of new cooling structure mock-up
Heat flux W thickness W surface temperature(°C) Cooling water velocity
15MW/m2 2mm 652 3.2m/s
5mm 798
20MW/m2 2mm 842
5mm 1101
The heat sink is CuCrZr and 316L stainless steel bonded structure by explosion welding technology. Before bond cooling channels was machining on CuCrZr side, and all the channels filled by some special metal. After CuCrZr plate and SS-316L plate explosion weld together the filled metal clean up both by higher temperature and chemical method from water inlet/outlet. Tungsten slices bond to CuCrZr plan apply braze, HIP and surface nanocrystallization + pressure under ~400°C with pure copper as inter layer. Braze and HIP already became mature technology, but the third technology is a new one. Compare three kinds of technology braze need high temperature (~ 960°C) will decrease CuCrZr properties. Cost of HIP is quite high. The third one can avoid CuCrZr degradation and reduce cost.
Mock-ups with special cooling channel and 2mm tungsten brazed to CuCrZr have been made and high heat flux tested in electron beam heating facility. The results shows as table 2. Mock-ups of tungsten bonded by HIPing and the third technology is under developing. High heat flux test will be done soon. The presentation will compare advantages of different technologies.
Table 2. High heat flux test results
Heat Flux W thickness W surface temperature(°C) Cooling water velocity
15MW/m2 2mm 780 3.2m/s
20MW/m2 970

[73] Modelling of cooling performance in single and multi-channel high heat flux structures for fusion applications

Presenter: SHARP, Samuel (Loughborough University)

A numerical study is conducted to explore the thermal efficiency of cooling streams in geometries relevant for fusion reactor high heat flux components. A tile-type structure is considered employing one or more cooling channels within the heatsink, based on recent investigations showing the benefits of this approach and advances in additive manufacturing. Various flow configurations are explored and compared to each other in their potential to extract and distribute the strong heat flux from the plasma-facing surface, differing for i) type of fluid; ii) inlet turbulence level; iii) velocity profile (swirled or unswirled); and iv) configuration (single or multi-channel). The study is conducted using a fluid-structure interaction solver implemented in the commercial code StarCCM. A conjugate heat transfer model is used for the structural analysis and an unsteady-RANS model with k-omega turbulence closure is used for the flow. The configurations are varied keeping the same mass flow rate and fluid-structure interface area. It is worth noting that this analysis focuses prevalently on the flow behaviour, while the structural analysis is used with the only purpose to provide a meaningful temperature (or energy equivalently) boundary condition on the fluid-structure interface, which would need to be approximated otherwise.

The analysis indicates advantages of multi-channel configurations over single-channel, consistent with previous work. Furthermore, the potential benefits of using swirled configurations with certain levels of turbulence are shown, and how these can be achieved without penalties in pressure drops is discussed from both flow and manufacturing points of view. Further considerations on the flow and heat transfer behaviours within and near the boundary layer are given in the paper.

[28] Simulation study of the radiative quasi-snowflake divertor for CFETR

Presenter: YE, Minyou (University of Science and Technology of China)

In the future fusion reactor, huge power has to be exhausted through the divertor due to the high fusion power. Therefore, it is critical to find an appropriate way to reduce the heat load onto divertor target, which has an engineering limit of 10 MW/m2. In snowflake configuration [1], second order null is introduced to increase the flux expansion as well as connection length, which benefits the wetted area and the radiation volume. However, due to the limit on the ability of the coils, in China Fusion Engineering Test Reactor (CFETR), it is hard to achieve the exact/near-exact snowflake configuration. Instead, by introducing additional divertor coil, a quasi-snowflake configuration can be achieved, where the flux expansion can be increased by a factor of ~2 compared with the single-null configuration, even the second null is still at a distance from the X point [2]. On the other hand, it is also an effective way to dissipate the heat power in the scrape-off layer (SOL) by radiation. For CFETR, due to the consideration of tritium retention and wall sputtering issues, full tungsten wall is proposed, which means impurities of high radiation efficiency have to be seeded to form a radiative divertor. It is natural to combine the quasi-snowflake configuration and impurity radiation together to find the effective solution of power exhaust. However, it is still not well understood the influence of flux expansion on the divertor physics, such as detachment, impurity screening, and so on. Simulation study is performed for the radiative quasi-snowflake divertor using SOLPS [3]. By varying the deuterium and impurity puffing rate, the boundary plasma with different radiation fraction is simulated for different upstream density. It is found, for the low density case, due to the influence of flux expansion in quasi-snowflake divertor, the outer divertor is detached earlier than the inner divertor, together with a strong impurity compression in the outer divertor. For the high density case, the detachment is more symmetry for inner and outer divertor.

Reference

[1] D.D. Ryutov, Phys. Plasma 14 (2007) 064502.
[2] S.F Mao et al., J. Nucl. Mater. 463 (2015) 1233.
[3] M.Y. Ye et al., Nucl. Fusion, doi: 10.1088/1741-4326/ab2bd0, 2019.

[38] A multi-physics modeling approach to predicting erosion, re-deposition and gas retention in fusion tokamak divertors

Presenter: WIRTH, Brian (University of Tennessee, Knoxville)

Plasma-surface interactions (PSI) span diverse physical processes as well as many decades of time and length scales (ps–s and Å–m). Correspondingly, comprehensive modeling of PSI must accurately target each scale and mechanism. Here, we present an integrated model designed to capture the multi-physics nature of interactions between the edge plasma and the divertor surfaces in a fusion tokamak. This workflow includes SOLPS simulations of the edge plasma in steady-state conditions; the effect of the sheath at shallow magnetic angles, evaluated by hPIC; GITR calculations of migration and re-deposition of impurities eroded from the divertor surface; and the divertor response to these plasma conditions, which includes evaluating surface growth and erosion, as well as sub-surface gas dynamics, modeled by coupling F-TRIDYN and Xolotl. We benchmark this workflow against dedicated PISCES experiments, which measured mass loss, spectroscopy and gas concentration profiles for W substrates exposed to mixed (D-He) plasmas. Given the positive comparison, we apply the model to predicting impurity migration and re-deposition, surface growth and erosion, and gas recycling in the ITER divertor, under conditions expected for helium and burning-plasma operations.

SOLPS predicts standard strongly radiating, partially-detached edge plasma in the divertor for both ITER scenarios. Our model shows that during He plasma discharge much of the impact energy-angle distributions are below the energy threshold for W sputtering. However, the high-energy tails of He+ and He2+ extend well above this threshold, leading to net erosion across the outboard divertor target, and despite the strong W re-deposition predicted by GITR. Xolotl predicts that the surface position reflects the balance between this erosion/re-deposition of W, and swelling driven by He implantation, resulting in surface recession far from strike point, but growth near it (R-Rsep~0–0.15m). He retention is largest where plasma temperature is high due to deeper gas implantation, even though the flux is ~10x less than its peak value. Under burning plasma conditions, Ne is the main radiative species and main contributor to wall erosion. Over 90% of the eroded W locally re-deposits, and produces net deposition where the plasma temperature is low (R~Rsep<0.15m), and to net erosion where the plasma temperature is high (R-Rsep>0.2m). The depth profiles of gases implanted in the W divertor are not strongly impacted by dilute impurities, at least over 10 second operating times. However, heat fluxes greatly affect the sub-surface D and T profiles, as increases in substrate temperature (from 343K to 525K at the peak heat flux location) during steady-state operation lead to faster gas diffusion, both into the bulk and outgassing. We also demonstrate our integrated PSI modeling capability by evaluating the influence of pre-exposure of the W substrate to He plasma (e.g., from the He-operation) on the tungsten divertor response to burning plasma operation. In this case, the higher concentration of He and vacancy clusters near the surface locally increase the D and T concentration (relative to an initial crystalline W) and reduce the permeation of hydrogenic species.

[32] A Study of the Maintainability of the Lower (Divertor) Port & Divertor Cassette

Presenter: WILDE, Andrew (UKAEA)

The EU DEMO project aims to prove that fusion power can be developed into a commercially viable power source. To achieve this, the plant needs to both produce sufficient power and be able to utilise it, and to demonstrate a closed tritium cycle. This requires that the plant needs to have suitably high availability that can be extrapolated to a commercial power plant (i.e. plant downtime for maintenance needs to be minimised) whilst meeting the relevant safety standards (IAEA, regulator, and good practice). Thus, maintainability becomes mission critical for DEMO.
This requires a step change in the level of integration of maintenance into the design efforts, compared to previous fusion experiments, from component level upwards to plant layout and design, with maintenance aspects becoming design defining.
The DEMO divertor cassette has evolved through several differing baselines and will continue to do so as is expected in pre-conceptual studies. During the course of this evolution, the engagement between integration, component design and maintenance teams has led to various lessons being learned and the development of design features which are required in the divertor cassette, along with several key learnings as to the effects of design decisions on the maintainability of the divertor cassettes and the impact on the lower port in general.
In this contribution we describe some of the handling and maintenance strategies which have evolved alongside the developing baselines, and describe the lessons learned which are likely to be common with future configurations of the divertor cassette.

[56] Investigation of detachment in Double-Null configurations in the TCV tokamak

Presenter: FÉVRIER, Olivier (Ecole Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-1015 Lausanne, Switzerland)

Plasma exhaust is a crucial issue for future fusion reactors. The high-power level across the separatrix needed to ensure H-mode operation and the narrow Scrape-off Layer (SOL) width make the task of staying within acceptable target heat loads extremely challenging, probably necessitating operation in a detached regime. In the past few years, significant efforts have been devoted towards the development of advanced divertor configurations that could facilitate access to such a regime. In particular, the double-null (DN) configurations may be an interesting and promising candidate. By magnetically separating the outer and the inner legs, DN configurations may allow the exhausted power to be shared between two outer legs, and, possibly, at two radiation fronts. Advanced geometries can then be applied to both active legs, together or separately, to increase the expected benefits.
In this work, we investigate detachment on TCV in DN geometries for a range of outer leg positions, including a double Super-X configuration, where the two outer strike points are located at high major radius. Measurements of the CIII emission front position along the lower outer leg show a movement at the front towards the X-Point at a lower line-integrated density $\langle n_e\rangle$ in the DN configurations, as compared to equivalent Lower Single Nulls (LSN), indicating a lower detachment threshold. This is further supported by Langmuir probe measurements at both outer strike-points. Bolometric measurements indicate that, for the same $\langle n_e\rangle$, a higher fraction (10% - 50%, depending on shape and $\langle n_e\rangle$) of the input power is radiated in these DN configurations. However, this enhanced accessibility of the detached regime appears to come at the price of a reduced detachment window. The double-null configurations disrupt at lower (between 10% - 20%) line-averaged densities than the equivalent LSN, after a swifter movement of the CIII front towards the X-Point. The sensitivity of these results to the magnetic balance of the double-null will also be discussed.

[17] Thermal hydraulic modeling and analysis of ITER tungsten divertor mono block

Presenter: EL-MORSHEDY, Salah El-Din (Prof. Dr. of Thermal-hydraulics, Egyptian Atomic Energy Authority)

The divertor is a fundamental component of fusion power plants, being primarily responsible for power exhaust and impurity removal via guided plasma exhaust. Due to its position and functions, the divertor has to sustain very high heat flux arising from the plasma (up to 20 MW/m2), while experiencing an intense nuclear deposited power, which could jeopardize its structure and limit its lifetime. Therefore, attention has to be paid to the thermal-hydraulic design of its cooling system. In this work a mathematical model has been developed/updated to investigate the steady state and transient thermal-hydraulic performance of ITER tungsten divertor mono block. The model predicts the thermal response of the divertor structural materials and coolant channel. The selected heat transfer correlations cover all operating conditions of ITER under both normal and off-normal situations. The model also accounts for the melting, vaporization, and solidification of the armour material. The model divides the coolant channels into a specified axial regions and the divertor plate into a specified radial zones, then a two-dimensional heat conduction calculation is created to predict the temperature distribution for both steady and transient states. The model is verified against a previous calculation in the literature for DEMO divertor at an incident surface heat flux of 10 MW/m2. The model is then used to predict the steady state thermal behaviour of the divertor under incident surface heat fluxes ranges from 2 to 20 MW/m2 for a bare cooling tube and a cooling tube with swirl-tap insertion. The model calculates maximum tube surface heat flux and the minimum critical heat flux ratio for all cases as well. The model is also used to simulate the divertor materials response subjected to high heat flux during a vertical displacement event (VDE) where 60 MJ/m2 plasma energy is deposited over 500 ms.

[5] Activity and Decay Heat Estimates for the European DEMO Divertor with Respect to WCLL and HCPB Breeder Blanket Module Integration

Presenter: TIDIKAS, Andrius (Lithuanian Energy Institute)

This work was carried out within the framework of EUROfusion/PPPT SAE (Safety and Environment) project. Activity and decay heat values were calculated for the DEMOnstration power plant (DEMO) 2015 baseline model divertor. Two irradiation scenarios were considered lasting for 5.2 and 14.8 calendar years respectively. Each irradiation scenario describes continuous irradiation with exception of 10 days in the end where irradiation corresponds to an operation cycle (4 hours of irradiation, 1 hour of rest). Activation characteristics for divertor were obtained with regards to three different blanket module configurations: single-module segmentation water cooled lithium lead (WCLL SMS), multi-module segmentation water cooled lithium lead (WCLL MMS) and helium cooled pebble bed (HCPB). Divertor model consists of 62 segments subdivided into 4 layers with different material makeup. Neutron transport calculations were performed with MCNP code with JEFF-3.2 nuclear data library. Activation calculations were performed with FISPACT code with EAF-2010 nuclear data library.

In general, lowest decay heat and activity values for whole divertor were seen in HCPB model configuration followed by WCLL MMS and WCLL SMS.

[49] DEMO Divertor - Cassette Design and Integration

Presenter: MAZZONE, Giuseppe (ENEA Department of Fusion and Technology for Nuclear Safety and Security, via E. Fermi 45, 00044 Frascati, Italy)

The divertor is one of the key components of the EU-DEMO reactor. The development of a reliable solution for the power and impurity particle exhaust is recognized as a major challenge towards the realization of DEMO. The pre-conceptual design activities for the EU-DEMO divertor are carried on considering two project areas: the ‘Target development’, focusing on the design of the vertical targets directly facing the plasma, and the ‘Cassette design and integration’, dealing with the design of the cassette structure and the integration of sub-components. The essential aim of the project is to develop in both project areas advanced design concepts for a divertor system being capable of meeting the physical and system requirements defined for the EU-DEMO reactor.

In this work a general overview of the EU-DEMO divertor cassette design is presented, considering systems & functional requirements, structural assessments and interfacing systems. The design solutions adopted for the integration of the main divertor sub-components are discussed, in terms of layout and attachment to the cassette body (CB) of the Plasma Facing Components (PFCs), the liner, the reflector plates and the cassette-to-vacuum vessel fixation system (nose at the inboard and wishbone at the outboard).

Different materials are integrated on the divertor cassette, requiring different cooling temperatures and leading to different behaviors to consider. In particular, Eurofer97 ferritic-martensitic steel has been selected for the cassette structure in order to meet the activation and radwaste requirements. As a consequence of this choice, two different cooling circuits have been introduced on the divertor cassette, one working at 180°C at 3.5 MPa as operating conditions of the Eurofer CB, the other providing coolant water for the PFCs (CuCrZr pipes) at 130°C at 5 MPa.

The main issues driving to the divertor design are tackled, mainly in terms of material damage for different materials present in the divertor assembly, VV shielding function of the divertor, optimization of both Vertical targets and Cassette body cooling circuits, dimension and position of opening on the divertor assembly necessary for pumping deuterium and helium from the lower port.

[7] Radiation-condensation instability: a driver for up-down or in-out asymmetry of divertor plasma

Presenter: KUKUSHKIN, Andrei (NRC Kurchatov Institute)

A spontaneous break of the up-down symmetry of the divertor plasma parameters in a symmetric transport model (symmetric double-null divertor configuration and boundary conditions, as well as the absence of drifts) in the presence of impurity seeding was found in computational modelling [1], [2]. The effect was attributed to the radiation-condensation instability (RCI) that amplifies the perturbations of the plasma temperature and drives the radiating impurity to the colder divertor. The behavior of the seeded impurity in a single-null divertor sometimes resembles this pattern, showing sharp re-distribution of the radiation intensity between the inner and outer divertors [3].

In the present paper, development of the RCI in the SOL and the divertors that it connects is studied by means of SOLPS4.3 [4] modeling. The code is set up to describe the parallel transport only (a 1D approximation) in order to simplify the model and to exclude possible interference of mechanisms other than RCI. The results show that the RCI can be the principal mechanism driving the asymmetries in the SOL and divertor plasmas.

[1] A. S. Kukushkin, “Spontaneous Break of up-down Symmetry in a Symmetric Double-Null Divertor Configuration,” Plasma Phys. Reports, vol. accepted, 2019.
[2] A. S. Kukushkin and S. I. Krasheninnikov, “Bifurcations and oscillations in divertor plasma,” Plasma Phys. Control. Fusion, vol. 61, no. 7, p. 74001 (8pp), 2019.
[3] H. D. Pacher et al., “Impurity seeding and scaling of edge parameters in ITER,” J. Nucl. Mater., vol. 390–391, pp. 259–262, 2009.
[4] A. S. Kukushkin, H. D. Pacher, V. Kotov, G. W. Pacher, and D. Reiter, “Finalizing the ITER divertor design: The key role of SOLPS modeling,” Fusion Eng. Des., vol. 86, no. 12, pp. 2865–2873, 2011.

[22] Some implications of recent technology advances on divertor physics performance requirements of DT fusion tokamaks

Presenter: WISCHMEIER, Marco (IPP Garching)

The continuing rapid evolution of a number of advanced technologies being strongly pursued for major non-fusion applications, is potentially transformative for the divertor physics performance requirements of reactors:
Advanced Manufacture, e.g. 3D printing, holds promise to increase the power handling capability of solid divertor targets significantly above the present limit for total power deposited on the target, ~ 10 MW/m2, by increasing the contact area between coolant and solid surfaces, and by reducing the risk for failure from fatigue.
Robotics. When the toroidal field coils are not openable, as in ITER, then all internal components including the divertor must be modular for construction, maintenance and repair since the components have to be moved in and out of the vessel through ports. Advances in robotics can be expected to enable improved installation, maintenance and repair procedures for modular devices, making it possible to use smaller gaps between modules/tiles and to reduce radial misalignments. This will be highly beneficial: in order to shadow-protect leading edges, the relatively large gaps and misalignments in ITER necessitate that the angle between B and the target surface, θ_, be restricted to very large values, ~ 4.5o, compared to typical values that can be safely used in present tokamaks, ~ 1-2o. This then causes much higher loads on the primary power-handling surfaces of the targets than would occur if smaller θ_could be safely used.
HTSC magnets. High temperature superconductor toroidal field magnets have the potential of being openable which makes it possible to use monolithic rather than modular internal structure of the vessel. ARC, for example, calls for HTSC toroidal field coils and θ_ ~ 1o; FNSF-AT, although using copper magnets, also calls for openable coils and θ_ ~ 1o. For both devices the entire, highly-aligned and structurally-strong, monolithic divertor would be pre-assembled and lowered in and out of place, thus making possible the safe use of small θ_.

These technological advances have potentially major implications for a number of critical divertor physics performance requirements regarding survival of the targets in reactors:
optimal plasma temperature and density at the divertor target, Tt¬, nt,
minimum level of volumetric power dissipation in the SOL/divertor, P_(diss-edge),
values of upstream plasma density (at the outside midplane separatrix).
R&D in these high technology areas is being pursued for major and rapidly growing non-fusion applications. The robotics industry, for example, is doubling in size every ~3 years. If these advances are proven at the time that a fusion power device is being designed, it will be essential that they be fully exploited. While the ITER design is now largely fixed, it will be possible to take advantage of advances in robotics and additive manufacture to upgrade the ITER divertor, which is designed to be replaced.

[51] Overview of the gas baffle effects on TCV Lower Single Null edge plasmas: multi-code simulations and comparison with experiments

Presenter: GALASSI, Davide

A gas baffle is being installed in the vessel of the tokamak à configuration variable (TCV) [1], in order to improve the closure of the divertor region. This upgrade has been envisaged, along with a foreseen increase in the available input power, in order to facilitate the access to detached divertor regimes at lower plasma collisionality, namely in more ITER-relevant conditions. It is necessary, in this framework, to be able to predict the impact of gas baffles, at the same time validating the current numerical tools available for the simulations of the edge plasma.

The design of the gas baffle tiles has been supported by SOLPS-ITER simulations of TCV edge plasmas [2]. The optimized parameter is the neutral compression ratio, namely the ratio of neutral density in the divertor region and the one in the main chamber, at a given upstream electron density. The performance of the gas baffles in terms of neutral confinement has been predicted for an ideal case with intermediate plasma current, as a function of plasma density and input power [3]. These simulations predict an improvement of the compression ratio, with the current baffles, of a factor of approximately five for attached low-density plasmas, which increases to up to 10-20 at higher upstream densities. Envisaging a future modification of TCV gas baffles, a further numerical investigation on the optimal baffle extension has been carried out, by means of the SolEdge2D-EIRENE code [4]. The main advantage of this code is the possibility to simulate realistic shapes for plasma-facing components, thus coherently describing plasma parallel fluxes impinging on gas baffles. This analysis shows that the neutral compression factor in detached conditions could be further improved by almost a factor two by extending the Low-Field Side baffle by a few cm. With such a solution, the recycling on the baffle tip would still be acceptable.

Both the mentioned numerical tools have been tested against experiments, SOLPS-ITER simulating an average current, Lower Single-Null scenario, and SolEdge2D-EIRENE a case at lower plasma current. Numerical results have been compared to experiments with baffle-compatible plasmas, in absence of the baffle. The density ramp imposed in experiments is reproduced numerically, and it allows the investigation of different divertor conditions. Transport coefficients are chosen to match experimental upstream profiles: numerical results are shown to reproduce, almost in a quantitative way, the experimental results at divertor targets. Keeping transport coefficients fixed, two additional wall geometries have been simulated, one including the High-Field Side baffle, and one with the full gas baffle. With the first experiments in baffled operation, both mentioned tools are tested against experiments: the results of the comparison will be discussed, shedding light on the capability of 2D transport codes of predicting the edge plasma behaviour in presence of gas baffles.

References

[1] H. Reimerdes et al., Nucl. Mater. Energy 12 (2017) 1106-1111.
[2] A. Fasoli et al., Nucl. Fusion (2019), submitted.
[3] M. Wensing et al., Plasma Phys. Contr. Fusion 61 (2019) 085029.
[4] H. Bufferand, et al., Nucl. Fusion 55 (2015) 053025.

[63] The Impact of Nonambipolar Energy Flow on Plasma Facing Materials Erosion and Forecast for ITER.

Presenters: KHIMCHENKO, Leonid (Institution "Project center ITER"), BUDAEV, Viacheslav (RNC "Kurchatov Institute")

In reactor-size fusion devices, such as ITER and DEMO, are expected the critical loads on the divertor plates both during steady state and at transient events (disruption, VDE, ELMs, runway electrons). High heat plasma load leads to enhanced erosion and destruction of material surface accompanied by enhanced absorption of tritium in erosion products.

The paper introduce the experimental dates of nonambipolar plasma flow, due to arcs and sparks, as mechanism of power exhaust, leading to extremely high heat load. The regimes with nonambipolar energy flow on tungsten limiter tiles obtained on the T-10 tokamak for quasistationary stage [1]. In such regimes the interior part of the tungsten limiter is heated up to temperature exceeded 3000 0C and estimated load is more than 50 MW/m2. Sparking, powerful arcing, deep cracks, edge melting and melt motion on ITER-grade tungsten tiles were observed. Also, tiles surfaces were flood by recrystallized tungsten. The edges of the cracks were melted and much arc craters have been located along the cracks.

The nonambipolar mechanism of energy flow on metal surfaces and self-heating, accompanied by sparks and arcs, is discussed to explain enhanced heating of PFM. Additional energy flow from plasma to the metal surface caused by the phenomenon of explosive electron emission (EEE), during the sparks activity [2]. The sparks accompanied by a continuous renewal of microexplosions, which initiated by the plasma and jets of liquid metal from previous microexplosions. Unlike “classical” thermoionic emission, such mechanism can increase electron flux into the plasma over the order of magnitude. It can change the energy and particle balance in the periphery plasma of a fusion reactor and leads to significant heating and melting of divertor plates and, accordingly, to cracking and melt motion. The reason of such sub-μs discharges ignition can be plasma-turbulence-driven fluctuations of particle and energy flux to the plasma-modified surface.

The results of W-tiles erosion by powerful runaway electrons presented, for comparison.

The report analyzes consequences for ITER the EEE appearance on the divertor W surface - the sharpening of SOL power width distribution, parallel to the magnetic field – □q; the melting of the W leading edges of divertor targets and the recrystallization of the W surface as a result of the superheated liquid metal droplets appearance. Melt tungsten can be subject to J × B force. EEE can lead to the erosion enhancement of the divertor plates. Micro-explosions lead to droplets, which, like dust particles, can effectively deliver impurities to the central region of the plasma.

[1] L.N.Khimchenko et al, SOFE-2017, Shanghai, China, report #496.
[2] S.A.Barengolts, G.A.Mesyats, M.M.Tsventoukh 2010 Nucl. Fusion 50, 125004

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Tuesday 05 November 2019

Registration - Board Room C (C Building, 4th Floor) (5 Nov 2019, 08:30-09:00)

Arrival of participants, distribution of badges, possibility to register and pay for events.

Implications of Applied 3D Fields - Board Room C (C Building, 4th Floor) (5 Nov 2019, 09:00-10:10)

time	[id] title
09:00	**[77] Three-Dimensional Boundary Physics Aspects for the Development of Next Generation Divertor Concepts with Resonant Magnetic Perturbations** *Presenter: JAKUBOWSKI, Marcin (Max-Planck-Institut für Plasmaphysik)* Resonant magnetic perturbation (RMP) fields applied for control of edge-localized modes (ELMs) break the axisymmetry of the plasma boundary in tokamaks. With RMP fields applied, a striation of the divertor target heat and particle flux pattern is detected which proves existences of helical magnetic fingers reaching from the X-point outward to the divertor target. A threedimensional plasma boundary is formed, which potentially alters the divertor and plasma edge transport characteristics. In this contribution, a systematic assessment of the heat and particle flux pattern with RMP fields for various tokamaks around the world is presented and discussed with respect to future divertor scenarios expected for ITER and DEMO. Main points of the discussion will be (a) the 3D plasma transport, (b) indications for the relation to the plasma response and (c) the impact on the actual divertor recycling condition including scaling into detached divertor regimes. The discussion will show that several open points exist to enhance the reliability of our capacity to extrapolate. The material will be presented to aid an subsequent open discussion of the relevance and impact these aspects for future devices. Acknowledgement: This work was conducted as part of ITPA-DSOL task 37. As such, many colleagues from various devices will be involved, which will be listed as co-authors of the presentation if the contribution is selected.
09:30	**[80] Recent progress in understanding the outer divertor heat flux dynamics during the ELM-crash-suppression by RMPs on KSTAR** *Presenter: LEE, Hyungho (National Fusion Research Institute)* For the reactor-scale fusion devices such as ITER or DEMO, control of the divertor target power loading, both in steady state and during ELMs, is particularly challenging with regard to tungsten target lifetime. It should be preferably reduced below a certain value so that the divertor target cooling capability ensures a planned long-term replacement period of the targets. It is widely accepted that resonant magnetic perturbations (RMPs) can be an effective method to achieve this since the peak heat flux is expected to be substantially reduced due to the profile broadening while ELM-crash being suppressed or significantly mitigated. Since the installation of a new, high spatial resolution outer target IR thermography system [1], the characterization and control of the outer divertor heat flux during the application of RMPs has been one of leading research subjects on KSTAR. In KSTAR, 3-row and 4 column In-Vessel Control Coils (IVCCs) are used to apply the magnetic perturbation of the toroidal mode number is 1 or 2 with various phasing and phase configurations. Especially, recently improved controllability of the RMPs configuration has made it possible to investigate the divertor heat flux dynamics according to various RMPs configuration during a single plasma discharge while keeping the ELM-crash suppression region. In KSTAR, it has been found that the divertor heat flux profile is clearly split showing the pattern likely what is expected by the field line tracing calculation according to the RMPs phases although it has been found that some details can be slightly different regarding plasma response models [2]. On the one hand, it has been observed that the outer target peak heat flux usually becomes much higher during the ELM-crash-suppression regime than that without RMPs. Since the observation seemingly contradicts the expectation from the EMC3-EIRENE calculation, the underlying physics in the phenomenon is under investigation on KSTAR. In addition, it has been demonstrated that intentionally misaligned ITER-like 3-row RMPs can not only suppress the ELM-crashes, but also disperse divertor heat fluxes in a wider area reducing the peak heat flux, while minimizing EM loads on RMP coils [3]. Along with the study on the RMPs configuration optimization, the methodology to obtain high density plasma ensuring RMP-ELM-suppressed regime compatible with detached divertor is ardently searched and some promising result has been achieved in KSTAR. In this paper, these recent findings, which has enhanced our understanding in the outer divertor heat flux dynamics during the ELM-crash-suppression by RMPs on KSTAR, are discussed compared to the simulation results by field line tracing or EMC3-EIRENE. [1] H. H. Lee et al., Nuclear Materials and Energy 12 (2017) 541. [2] H. H. Lee et al., 2nd IAEA Technical Meeting on Divertor concepts (Suzhou, China, 2017). [3] Y. In et al., Nuclear Fusion, submitted.

09:50 **[76] First-time analysis of detached divertor conditions in RMP ELM suppressed H-mode plasmas in ITER**

Presenter: FRERICHS, Heinke (University of Wisconsin - Madison)

The ITER divertor has been designed for axisymmetric configurations, yet symmetry breaking resonant magnetic perturbations (RMPs) will be applied for control of edge localized modes (ELMs). Recently,the numerical capability to investigate the predicted detached divertor scenario at ITER with such 3-D deformations has been made available after stabilization of the iterative framework [H. Frerichs et al., NME 18 (2019) 62] of the 3-D edge plasma and neutral gas code EMC3-EIRENE.

We have applied an n=3 RMP field to the baseline discharge for the pre-fusion power operation phase and included the plasma response from calculations with the single fluid resistive MHD code MARS-F. The divertor state is found to be sensitive to the toroidal flow impact on the plasma response. Even though screening of most of the resonances leads to a narrower region of broken flux surfaces, radial extension of the divertor footprint occurs due to field amplification near the separatrix. Detachment transition with RMP occurs at a lower gas puff rate and lower peak particle flux at the original strike zone, consistent with a lower upstream heat flux that it connects to along the 3-D scrape-off layer. However, a secondary non-axisymmetric strike location exists radially further outward, which remains attached because of a magnetic connection to higher upstream temperatures further inside the bulk plasma, carrying significant heat fluxes to this previously low flux domain in the divertor. This strongly reduces the potential for complete power dissipation, and it is a new feature in the ITER divertor revealing a challenge for the integration of RMP ELM control with a feasible divertor operation regime. The new results will be put into context with recent modeling efforts at other machines that include plasma response effects on the boundary plasma [J.D. Lore et al., Nucl. Fusion 57 (2017) 056025, M. Faitsch et al., 2019 PPCF 61 014008].

This work was supported by the US DOE under DE-SC0012315 and DE-SC0013911, by the CoE at the UW - Madison, and by the ITER Science Fellow Network.

Coffee Break - Board Room C (C Building - 4th Floor) (10:10-10:30)

Discussion Session: 3D Fields - Board Room C (C Building, 4th Floor) (5 Nov 2019, 10:30-11:10)

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

Stellarators Divertors - Board Room C (C Building, 4th Floor) (5 Nov 2019, 11:10-12:30)

time [id] title

11:10 **[31] The Island Divertor Concept of the Wendelstein 7 Stellarator Line – Concept, Experimental Experience and Up-scaling to Reactor Relevant Size**

Presenter: KOENIG, Ralf (Max-Planck-Institut for Plasma Physics)

The island divertor (ID) concept investigated experimentally in the Wendelstein 7-AS (W7-AS) and the superconducting Wendelstein 7–X (W7-X) stellarator devices have so far proven to be extremely successful, and shown a favourable tendency towards improved detachment performance from W7-AS to W7-X. Stable detachment, which could be achieved only partially in W7-AS and required additional resonant magnetic perturbation (RMP) fields, is more easily accessible in W7-X without the necessary need of external RMP fields. The W7-X detachment is usually more complete in the sense that the target heat load reduces quite homogeneously without leaving noticeable hotspots, unlike in W7-AS. In the last W7-X experimental campaign, a 5.5 MW ECR-heated high line integrated density (1.1 E20 m-2) plasma was stably maintained under completely detached conditions over 26 s without any signs of impurity accumulation (constant Zeff = 1.5). An order of magnitude reduction of the peak heat loads was observed on the target plates monitored by IR-cameras without any indications for remaining hotspots. The sub-divertor neutral pressures were sufficient to pump towards the end of the discharge just about as much as was fuelled. This was achieved in an uncooled test divertor, which is presently being replaced by a fully actively cooled High Heat Flux (HHF) divertor of identical shape, aiming at 30 min 10 MW ECR-heated plasma operation.

The ID concept utilises the inherent resonant radial magnetic field components of the 3D shaped coils of these devices, which create magnetic islands at the edge. The edge islands provide a natural separatrix configuration. These structures are intersected by 3D shaped target plates suitably adapted to the basic geometry and symmetry of the magnetic field configurations. The small field line pitch, i.e. the internal rotational transform around the island centre, leads to long connection lengths Lc (several 100 m) and thereby increases the relative weight of perpendicular transport. Besides Lc, the radial island width Wi is another important geometric parameter for the ID, which has been evidenced in W7-AS experiments. By making use of the in-vessel control coils and the external planar coils, it is possible to vary Wi and Lc independently from each other and to optimize the ID performance with respect to particle and power exhaust as well as impurity retention.

EMC3-Eirene simulations suggest that the, compared to W7-AS 2x larger islands, favouring access to high recycling conditions, and the more symmetric, drift optimised flux surface geometry are the main reasons for the improved detachment performance of W7-X. Indeed 2.5 times the separatrix density (5.5 E19 m-3) has been spectroscopically measured near the divertor target plates (1.3 E20 m-3) just before the transition into detachment. Based on the W7-AS and W7–X results, an estimate of the basic ID behaviour has been made for a device up-scaled from W7-X to a reactor scale device. This paper summarizes the main results obtained so far from W7-AS and W7-X and provides a brief physical interpretation of the leading effects and a rough assessment of their relevance to a reactor.

11:30 **[39] Advantage and disadvantage of the LHD heliotron divertor**

Presenter: KOBAYASHI, Masahiro (NIFS)

This paper discusses advantage and disadvantage of the LHD heliotron divertor in terms of divertor functions, such as neutral compression, impurity transport, and divertor heat load control.
The divertor plasma density in heliotron divertor stays at low values, ~1x10^19 m^-3, which never exceeds upstream density, i.e. at the stochastic layer, where ne can reach up to ~10^20 m^-3. This is interpreted as due to the strong magnetic shear at the edge stochastic layer, which squeezes and mixes up field lines of different connection lengths, and thus of different plasma pressures. The pressure loss through the enhanced cross-field interaction between the flux tubes results in low neutral compression in divertor region, ~0.1 Pa, even in closed divertor configuration. In order to compensate the low pressure, new cryo-sorption pump has been developed with an advanced technique on cryo-panels, which enables flexible shaping and compactness of the pumping system that can be installed directly under the dome structure in complex 3D shape. Pumping speed of 96.5 m3/s and capacity of 86,000 Pa m^-3 have been achieved so far, which can provide particle exhaust for low to medium density operations in LHD. Further upgrade for high density operation, ~10^20 m^-3, remains for future task.
The stochastic layer is distributed around the confinement region in all poloidal angles. Therefore, it provides effective screening against impurities coming from either divertor or first wall. VUV spectroscopy measurements with absolute calibration showed reasonably small impurity content in core plasma both for iron (~10^15 m^-3), which comes from the first wall, and for carbon (~10^17 m^-3), which comes from the divertor plates. EMC3-EIRENE simulations revealed clearly different features of impurity screening mechanisms between heliotron and tokamak configurations.
The divertor heat load is found to be strongly non-uniform along the helical strike lines, which overrides expected widening of wetted area due to the multiple strike lines and due to the double null divertor legs. On the other hand, because of the high density at the stochastic layer than the divertor region as described above, impurity radiation at the stochastic layer is main volumetric energy loss. Since Te at the stochastic layer changes from ~30 eV at the periphery to 300-500 eV at LCFS, this region has potential to provide various impurity line emissions. It is found that mixed impurity seeding of Ne and Kr, rather than single species, realizes stable radiative divertor operation and effective cooling of divertor plasmas. Control of edge radiation by RMP application has been also demonstrated, where radiation is enhanced around the magnetic island induced by the RMP. While these schemes are found effective to mitigate the divertor heat load, toroidal asymmetry in the heat load pattern appears during the detachment in certain operations. Core transport analysis shows changes in heat transport coefficient profile in the attached and detached phases, while no significant confinement degradation is observed. Based on these results, prospect for further divertor optimization will be discussed.

11:50 **[13] Divertor and Exhaust Modelling in the Framework of a Systems Code for a Stellarator Power Plant**

Presenter: WARMER, Felix (Max Planck Institute for Plasma Physics)

The controlled particle and heat exhaust is one of the most challenging aspects towards the realisation of a commercial fusion power plant. However, despite this importance, it is difficult to extrapolate the expected divertor heat load for a future fusion power plant. In the tokamak community, credible models for the divertor heat load in DEMO have long been missing and still largely rely on empirical scalings, which were derived from a large database of tokamak experimental data. A number of very sophisticated codes exist, but require prohibitively large computing resources to cover a larger parameter space.
Consequently, it is important to develop tractable models at affordable computational cost to assess the divertor placement and heat loads for the next-generation of power generating fusion devices. It is essential to be able to predict the expected heat load in order to ascertain whether the concept is actually feasible given the material limitations. In other words, the divertor heat loads may limit the design space, posing a strict boundary condition that needs to be taken into account during the conceptual design phase and optimisation of next-step devices. For this procedure, usually a Systems Code is employed that models an entire fusion plant. In this work we concentrate on the divertor and exhaust scenario for the stellarator concept.
The advanced modular 5-periodic stellarator concept currently favoured in Europe features a so-called magnetic island divertor concept. In this concept, the inherent magnetic field structure with a low order rational rotational transform at the plasma boundary and low magnetic shear leads to a chain of resonant magnetic islands in the plasma edge region, which provides a structured layer between the confined core plasma and the target elements. The divertor target plates are placed such that they intercept the islands to efficiently control the energy and particle exhaust.
Before the start of the Wendelstein 7-X stellarator, a heuristic island divertor model was developed for the Systems Code PROCESS and used for power plant design studies of the stellarator concept. In the meantime W7-X has become operational and demonstrated a robust particle and heat exhaust handling using the island divertor concept. The most notable achievement has been the stable and fully detached plasma operation for 30 seconds, which is a favourable scenario for a stellarator power plant.
Based on these results, the original heuristic island divertor systems code model is reviewed and validated against W7-X experimental results. Further, based on the experimental results, a way forward is devised for the development of an improved tractable island divertor model suited for Systems Code and design applications. Ultimately, this work aims to enhance the fidelity and credibility of the conceptual design studies of next-step stellarators addressing one of the most challenging aspects of fusion development.

Discussion Session: Stellarators - Board Room C (C Building, 4th Floor) (5 Nov 2019, 12:30-13:10)

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

Lunch - Board Room C (C Building - 4th Floor) (13:10-14:10)

Alternative Materials for PFCs - Board Room C (C Building, 4th Floor) (5 Nov 2019, 14:10-15:30)

time	[id] title
14:10	**[65] Fast Flowing Liquid Metal Divertor Design Options: Experimental and Numerical Studies**

Presenter: EGEMEN, Kolemen (Princeton University)

A fast-flowing liquid metal (e.g. Lithium, Tin) divertor (FFLMD) is an attractive option that can take all (or almost all) the heat flux coming to the PFCs. A generic fusion reactor divertor with “fast” flow generally requires a ~1-20 m/s speed with approximately mm to cm thickness. Balancing the heat flow into the divertor and carrying capacity of the liquid metal (LM) flow is the main requirement that sets the “fast” flow speed for the divertor. Such a divertor takes all the D/T particle flux and heat flux and allows substrate behind the liquid to be designed only for neutron fluxes. This permits the use of neutron-tolerant, low thermal conductivity, steels as substrates – an innovation which would greatly reduce material development for fusion. Among the challenges FFLMD concept need to overcome, the main one is the stabilization of the fast flow under MHD effects. In this presentation, we explain these effects and how to overcome them.

We present possible configurations including the fully toroidally connected annular FFLMD, toroidally segmented FFLMD, and arrangement of many “divertorlets,” small modular divertor systems. We explain how fast stable flow can be achieved in these systems and we explain the scaling laws for FFLMD technical requirements as the fusion reactor regime is approached. This theoretical analysis is then complemented with the experimental studies. We present the results and developments from the FFLMD experiments at Liquid Metal eXperiment, LMX, at PPPL and Liquid Metal FRee-surface EXperiment, LMFREX, that was placed in Oroshhi-2 in NIFS, Japan.

LMX studies the fast LM in a channel configuration with a magnetic field up to 0.33 Tesla and 2 m/s flow speeds. Heat transfer studies in LM found optimal channel surface shaping to obtain the maximum heat transfer from the surface that would minimize the evaporation in a reactor. The effect of surface shapes such as delta-wing and dimples at the bottom of the channel are shown experimentally and with numerical simulations. Running current through the LM for flow acceleration via magnetic propulsion and flow stabilization is studied experimentally. Analytical and numerical models of flow speed and wave variation are developed. This effect is used to control the hydraulic jump location.

LMFREX was placed in the Orosshi-2 facility to study LM flow in higher magnetic fields (3 Tesla). Under substantial vertical magnetic fields, MHD drag becomes a major problem for LM flow. We studied the effect of running poloidal currents, j, and showed that poloidal current can induce enough jxB force to overcome the drag and accelerate the LM to high speeds. This method can be used in a segmented LM divertor, which would allow running currents in toroidal direction, allowing FFLMD under vertical field conditions.

Finally, the next steps experiments that are needed to finalize the design of a FFLMD for reactor are discussed. FLIT, an upcoming torus device at PPPL that is designed to look at these issues under realistic conditions (1 Telsa and 10 m/s), is discussed briefly.

14:40 **[12] Liquid Metal Conceptual Divertor Designs for the European DEMO**

Presenter: MORGAN, Thomas (Dutch Institute for Fundamental Energy Research)

The crucial stepping stone between ITER and a fusion power plant is generally foreseen as a demonstration power plant (DEMO). The European approach foresees only a modest upscaling in dimensions from ITER but due to the large increase in fusion power and subsequently strongly increased power crossing the separatrix [1] this implies increased challenges for power exhaust. As a risk mitigation strategy alternative approaches to this issue are being pursued, including whether a liquid-metal (LM) based divertor could be an option for DEMO.

Such a divertor should be able to handle similar or greater heat fluxes to the baseline approach (an ITER-like W-monoblock based divertor [1]) but is attractive as it could show greater resilience against off-normal events and neutron loading leading to a more robust divertor. A set of design requirements to achieve this goal while conforming to the operational safety and fusion output requirements of DEMO have therefore been formulated in consultation with the European design team. Based on these requirements a series of conceptual designs have been developed within the EUROfusion workpackage WPDTT1-LMD. While several different approaches have been considered the leading candidates are water-cooled designs using tin as the liquid metal. FEM analysis shows that power handling capabilities well above 10 MW m-2 in steady state are achievable while conforming to design requirements. In addition slow transients, ELMs and disruptions appear tolerable without damage to the PFC. Tin confinement by mesh-based Capillary Porous Structure (CPS) is used, but novel approaches to its production, such as 3D printing, have been investigated. Modelling using TECXY and COREDIV shows that core concentrations of Sn can be limited to tolerable values by Ne or Ar impurity addition. Considerations such as wetting, corrosion and fuel retention are also being addressed. This contribution will discuss the design requirements, experimental inputs and modelling of the design and place it in the context of the European pre-conceptual design efforts for an LM-based divertor for DEMO.

[1] Federici, G. et al. Fus. Eng. Des. **109–111** (2016) 1464-1474.

15:10 **[20] Analyses and Experiments Towards a Lithium Vapor Box Divertor**

Presenter: GOLDSTON, Robert (Princeton University)

The divertor for a practical fusion power producing facility very likely must dissipate the intense heat flux emerging from the plasma core volumetrically, rather than allowing it to strike a material surface directly. We have proposed [1, 2] that a dense cloud of lithium vapor be contained in the divertor region by local evaporation from, and condensation onto, capillary porous structures such as 3-D printed tungsten surfaces [3]. Modeling has shown [4] that the heat flowing from a fusion-relevant plasma can be dissipated volumetrically by the radiation and ionization associated with encountering lithium vapor. It has been further shown that such a system can be designed to be robust against large variations in heat flux [5]. The very modest flows of lithium required for such a system can be easily pumped across magnetic fields [6]. Indeed capillary pressure alone is sufficient in the presence of flow channel inserts. Experiments are underway, and being developed, to test this concept in a stepwise manner. We are measuring the ability to contain a small cloud of lithium vapor consistent with calculations using the SPARTA direct simulation Monte-Carlo code. In parallel we are preparing the physics design of an experiment to test volumetric dissipation of a plasma beam on the Magnum-PSI facility [7] in such a localized lithium cloud. We are also preparing the pre-conceptual design of a lithium vapor box option for the divertor in EAST [8], and are developing plans for testing a full toroidal system at very high power density on the NSTX-U experiment. Such as system could also be tested in COMPASS-U and DTT.

[1] R. J. Goldston et al., Physica Scripta T167 (2016) 104017
[2] R. J. Goldston et al., Nuclear Materials and Energy 12 (2017) 1118
[3] P. Rindt et al., Nuclear Fusion 59 (2019) 054001
[4] T. D. Rognlien et al., Nuclear Materials and Energy 18 (2019) 233
[5] E. D. Emdee et al., Nuclear Materials and Energy 19 (2019) 244
[6] E. D. Emdee et al., accepted for publication in Nuclear Fusion
[7] J. A. Schwatz et al., Nuclear Materials and Energy 18 (2019) 350
[8] E. D. Emdee et al., European Physical Society, Division of Plasma Physics, 2019

Discussion Session: Alternative Surfaces - Board Room C (C Building, 4th Floor) (5 Nov 2019, 15:30-16:10)

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

Coffee Break - Board Room C (C Building - 4th Floor) (16:10-16:30)

Poster Session II - Board Room C (C Building - 4th Floor) (5 Nov 2019, 16:30-18:30)

All submissions accepted as "Poster" should provide a poster conforming to the rules published in the meeting announcement. Posters will be shown outside of Board Room C near by the coffee area.
Board numbers correspond to Indico [ID] numbers.

[id] title	board
[62] Impact of X-point geometry and neutrals recycling on edge plasma turbulence *Presenter: TAMAIN, Patrick (CEA Cadarache)* In tokamaks, heat and particle exhaust as well as confinement depend on the interplay of multi-physics phenomena occurring in the boundary of the plasma. A comprehensive modelling of the physics at play should involve a consistent description of plasma transport - including turbulence -, plasma-wall interaction, atomic and molecular physics, all treated in realistic magnetic and wall geometries. Due to the complexity of such modelling, the state of the art has for long been compartmentalized between mean-field codes, lacking a self-consistent description of transverse transport, and turbulence codes, ignoring neutrals physics and most often in simplified geometries. However, experimental results point out that the decoupling between turbulence physics and other aspects of edge plasma physics cannot properly account for observations. One can take as an illustration the difference between scrape-off layer (SOL) widths in limiter and divertor geometry [1,2] or the now well documented shoulder formation at high densities [3]. As part as an effort to bridge the gap between these two facets of edge plasma physics, we report recent results from edge plasma turbulence modelling with the TOKAM3X code. In a first part, we focus on the impact of X-point geometry on edge turbulence and transport. Our simulations exhibit a strong impact of the magnetic geometry. On the one hand, turbulent transport appears extremely sensitive to the magnetic flux expansion, leading to an absolute level of transport larger in areas of large flux expansion, typically at the top of the machine. The X-point itself is found to quench turbulence in its direct vicinity, leading to a quiescent zone along the outer separatrix below the X-point and a deconnection of turbulent filaments between the mid-plane and the divertor, thus impacting near SOL profiles and heat flux spreading in the divertor. In a second part, we present the results of first principle simulations including self-consistently turbulence and neutrals physics. For that purpose, TOKAM3X was coupled to the EIRENE kinetic neutrals code in order to account self-consistently for neutral particles recycling from the target plates and their interaction with the plasma. Comparison is made between simulations driven by a gas puff and including neutrals recycling and simulations classically driven by an arbitrary core particle influx. Results demonstrate that the change of location of the particle source from the core to the edge fundamentally changes the natures of heat transport in the closed field lines region. Non-linear interactions are also found between the neutrals and the plasma, leading to an amplification of poloidal asymmetries in the SOL, in particular an increase of intermittency in the immediate vicinity of the targets. [1] J. Gunn et al., J. Nucl. Mater. 438, S184-S188 (2013). [2] A. Scarabosio et al., J. Nucl. Mater. 438, S426–S430 (2013). [3] D. Carralero et al., Nuclear Materials and Energy, 1189-1193 (2017).	
[67] KINETIC TRAJECTORY SIMULATION METHOD FOR INTERACTION OF MAGNETIZED PLASMA HAVING TWO SPECIES OF POSITIVE IONS WITH TUNGSTEN SURFACE *Presenter: KHANAL, Raju (Tribhuvan University)* The kinetic trajectory simulation method has been employed to study the plasma-wall interaction in the magnetized plasma with two species of positive ions exposed to the tungsten (W)-surface. The multi-component plasma interacts with W-surface through non-neutral plasma sheath formed near the Plasma Facing Materials (PFMs). It is found that the ion velocity distribution functions have a cut-off Maxwellian distribution with almost equal magnitudes of cut-off and Maxwellian maximum velocities. The presheath electron temperature can significantly affect the wall potential and ion flow, whose explicit effect can be seen on the ion fluxes and current density at the wall. In addition, the reflected concentration of both the ions decreases so that absorption rate increases; however, the lighter ion absorption is about 7% higher in magnitude than that of heavier ions for the W-surface.	

[57] Tests of Plasma Facing Component Materials with Steady State Plasma

Presenter: BUDAEV, Viacheslav (NRU "MPEI")

Full-scale tests of the divertor materials like tungsten, lithium, tin components are extremely important to model relevant loads on plasma-facing materials of a fusion reactor. For such purposes, the PLM plasma device was constructed at NRU “MPEI”. The PLM plasma device is a linear system of a 8-pole multicusp magnetic field with parameters similar to the SOL plasma in a tokamak; stationary plasma heat load on test target samples is up to 5 MW/m2. The divertor tungsten mock-ups were irradiated with steady state plasma in the PLM. The combined tests of ITER-grade tungsten samples with e-beam load of 10–50 MW/m2 and stationary plasma load of 1–2 MW/m2 led to erosion, cracking, and nanostructured “fuzz” structure growth on the material surface. Capillary porous system of liquid tin was tested with stationary plasma in the PLM during ~200 minutes demonstrating sustainability to the high heat plasma load. Lithium materials deposited in the T-10 tokamak during experiments with lithium capillary-porous system were irradiated with stationary plasma in the PLM to test the evolution of the deposits under long-term plasma load.

[33] Integrating advanced plasma-wall interaction in 3D turbulent simulations for WEST

Presenter: BUFFERAND, Hugo (CEA)

In support to WEST operation, a dedicative effort has been made to improve 3D turbulent plasma simulations in particular to take neutral response and impurity sputtering into account. In this contribution, we present the latest results obtained with the code resulting from the merging of SOLEDGE2D and TOKAM3X which is able to cope with the realistic geometry of WEST plasma facing components thanks to immersed boundary conditions method. The plasma model used can address multi-species plasmas, following the collisional closure proposed by Zhdanov for multi-components plasmas. The plasma solver is also coupled to the neutral code EIRENE to take plasma recycling and wall sputtering into account. The new code can also be run in 2D in a transport code fashion where the turbulent transport is emulated by diffusive transport. A hybrid MPI-OpenMP implementation has been chosen to solve in parallel different plasma species and different geometrical subdomains.

First simulation results for WEST plasmas are presented and compared with experimental data.

[72] Impact of divertor configuration on tokamak performances: focus on WEST experiments supported by SOLEDGE2D modelling

Presenter: CIRAOLO, GUIDO (CEA, IRFM)

A matter of fundamental importance for future fusion reactors is the ability of combining a hot, high-performance core plasma with a cold plasma at the divertor plates, minimizing the heat loads onto the plasma facing components (PFC) and avoiding their erosion. These two regions are coupled by the Scrape-Off Layer (SOL), characterized by open field lines, which affects reactor performances. Indeed, it has been recently shown experimentally on JET that the divertor configuration can affect significantly the formation of the pedestal, lowering the L-H transition threshold by a factor of 2 [1], through the increase of the shear of the ExB flow in the SOL [2]. On the other side, recent analyses performed on Alcator C-mod indicate upstream plasma behavior relatively unaffected by strong changes in divertor conditions [3]. In order to progress in the understanding of the impact of divertor configuration and conditions on tokamak operation and performances we propose here further investigations based on the analysis of recent experiments on WEST tokamak supported by numerical simulations performed with the transport code SOLEDGE2D-EIRENE.

More precisely we focus on two fundamental parameters that have to be monitored during the operation: the electron temperature at the divertor plates Te,div and the pedestal pressure Pped. The aim is to minimize Te,div to values below a few eV, to avoid physical sputtering and erosion of the PFC, while maximizing Pped, determining the fusion power. In this contribution, we focus on the effect on Te,div and Pped of the misalignment between the parallel heat flux q// and the particle flux □ at the divertor plate. Indeed, a shift between the peaks of heat and particle fluxes is observed recurrently in recent WEST experiments, in agreement with predictions from SOLEDGE2D numerical simulations performed in preparation of WEST operation [4]. We investigate the impact of divertor configuration (single null vs upper null) as well as nitrogen seeding on such misalignment completing the experimental data with numerical results obtained with the transport code SOLEDGE2D-EIRENE.

[1] X. Litaudon et al., Nucl. Fusion 57 (2017) 102001
[2] E. Delabie et al., 42nd EPS Conference on Plasma Physics, Lisbon, Portugal (2015)
[3] B. LaBombard et al, oral presentation PSI conference, 2018
[4] G. Ciraolo et al., Nucl. Mat. En., 12 (2017) 187–192

[48] Radiative divertor experiments with Ne, N, and Kr seeding in LHD

Presenter: MUKAI, Kiyofumi (National Institute for Fusion Science)

This paper reports divertor heat load patterns and plasma responses observed in the Ne, N, and Kr seeding experiments in LHD.
The previous study showed that the edge stochastic magnetic field layer in LHD, where Te changes from ~30eV at the divertor legs to 300-500eV at the LCFS, provides the main radiation contribution rather than the divertor legs. Understanding and control of impurity radiation in this layer is, therefore, prerequisite for detachment operation in future helical devices. The magnetic field structure of LHD has 10 field periods in toroidal direction, and thus uniform heat load reduction in toroidal direction is important.
With Ne seeding, simultaneous reduction of the divertor heat load at all toroidal sections is not always successful, despite an increase in radiated power. Rather, at specific sections the heat load even increases after Ne seeding. The asymmetry lasts for 0.2sec, which is much longer than the transport time of charged impurity particles in one toroidal turn, and then the high heat load returns. The “specific sections” are not a simple function of distance from the impurity puff location, and change depending on the heating power and on the ne. After the Ne seeding Te at the divertor plates, measured by Langmuir probes, decreases down to 5-10eV at all sections. On the other hand, ne is found to increase at the “specific sections” with the increased heat load. Further increase of the Ne seeding level aiming at symmetric heat load reduction fails with radiation collapse.
The toroidal asymmetry is, however, effectively removed by the addition of Kr seeding prior to the Ne seeding. Kr emission remains very low until the Ne seeding, which then triggers Kr XIX to increase together with Ne VIII. Thereby, the divertor heat load decreases by a factor of ~2 in all toroidal sections. The reduction lasts for >1sec after the seeding. In this case, a clear decrease in Te at the stochastic layer is observed in the Thomson scattering measurements. Imaging bolometer measurements indicate a shift of the impurity radiation toward inner minor radii, i.e. around LCFS. Nevertheless, degradation of energy confinement time is limited to ~5% in the discharge. Kr seeding alone did not realize such operation. The results suggest a synergy effect between Ne and Kr. The relations between the magnetic field structure, radiation distribution, and the toroidal heat load pattern are under investigation.
N seeding discharges show stronger toroidal asymmetry than Ne seeding as expected, and also recovery of the divertor heat load after N seeding termination occurs very quickly, in several tens milliseconds. However, it is found that a slower seeding rate with a longer puff duration (>1sec) is effective to remove the toroidal asymmetry of the heat load. In this case, there is no significant degradation in the plasma stored energy observed.
These results suggest the importance of the mixture of impurity species as well as seeding rate to control the edge impurity radiation in the stochastic layer that appears in the 3D magnetic field structure of helical devices.

[71] Behaviour of Tin under Low-Temperature Deuterium Plasma Irradiation

Presenter: NEU, R. (MPI für Plasmaphysik)

Liquid metals have the potential to mitigate several issues inherent to solid divertor targets, e.g., problems arising from erosion, embrittlement due to neutron irradiation and crack formation under fast transient loads. As possible choice for such a liquid metal tin ($T_{melt} = 505$ K) was identified, which promises low physical sputtering yields and a large operational temperature range because of its low vapour pressure. For this reason, the behaviour of tin under deuterium plasma irradiation was systematically investigated at different target temperatures by exposing it to a well-characterized D plasma with an ion flux of $\approx 10^{20}$ Dm^{-2}s^{-1} at a bias voltage of -25 V. Since the sputter threshold of D on Sn is in the range of 70 eV the Sn erosion is expected to be negligible. However, as already indicated in the literature, Sn could form the metastable, volatile stannane (SnD$_4$) when exposed to D plasma. Although no stannane molecules where found in the exhaust gas, a large mass loss was found after exposure at 300 K which can only be explained by chemical erosion. At 495 K (i.e. 10 K below the melting point) the mass loss was strongly reduced by a factor of ten (compared with the exposure at 300 K). However at 515 K, in the liquid phase, the mass loss was dramatically increased. The latter increase is most probably due to the ejection of Sn micro-droplets, which were found on surrounding witness samples. These might be caused by bursting gas bubbles (see below). Depending on temperature and aggregation state (solid/liquid) large differences in the deuterium retention were found. Whereas the retention close to the surface (measured by Nuclear Reaction Analysis) is very high (> at. 1%) in the case of 300 K, it is at or below the detection limit (5x10^{-5} at. %) in the case of liquid Sn (515 K). However in all cases, bulk retention in the form of H bubbles is observed. Specifically at 495 K, a large sponge like structure reaching deep into the bulk (~100 µm) is observed. At 515 K, a large D2 bubble was formed in the crucible underneath the liquid Sn. It seems that the formation of metastable stannane could play an important role in all the above mentioned observations, but the detailed explanation and the consequences for the uses of Sn as a liquid plasma-facing material are not yet clear.

[46] Importance of divertor physics modeling in system design of LHD-type helical reactor

Presenter: KOBAYASHI, Masahiro (NIFS)

Helical systems inherently have a suitable feature as a future fusion power plant in terms of steady state operation because of no need of the plasma current drive. Among several configurations, conceptual design study of the LHD-type helical fusion reactor has been conducted and the design of the commercial scale power plant FFHR-d1, which can be operated with a fusion power of 3 GW, has been proposed.
Because there is a gap between the present knowledge and the requirements on future power plants in terms of both plasma physics and fusion engineering, the design parameters of the future fusion power plant will be changed by reflecting the latest knowledge from the plasma physics and engineering researches. Construction and operation of one or more intermediate machines are desired to ensure steady progress to the commercial power plant. In this respect, parametric scans of design parameters using a systems code are still important. On the other hand, calculation model of SOL and divertor plasma has not been implemented in the systems code for the LHD-type helical reactors. Because present LHD-type reactor design is a simple scale-up of the LHD, magnetic field line structure is also scaled by the reactor size. Assuming strong correlation between the divertor magnetic field structure and the divertor heat load distribution, the divertor heat load of LHD-type helical reactor is estimated from the total power to the SOL region and the divertor wetted area scaled from LHD. In the case of FFHR-d1 with a major radius of 15.6 m (4 times larger than LHD), the total divertor wetted area is estimated to be $\sim$32 m^2 from that of LHD ($\sim$2 m^2). The total power to the SOL region is estimated to be $\sim$550 MW considering the Bremsstrahlung loss of ~50 MW. Then average divertor heat load becomes $\sim$10 MW/m^2 assuming the 30% radiation cooling in the SOL region, which is not so difficult to achieve. However, divertor heat load has a strong inhomogeneity in the toroidal direction and local heat load can be several times higher than the average value, which cannot be accommodated by the conventional tungsten divertor. Then radiation cooling in SOL region and divertor detachment play an important role, and the modeling of these condition which is related to the design parameters handled in the systems code is required for the design window analysis by parametric scans of wide-ranged multiple design parameters.
In the presentation, the latest status of the systems code and system design of LHD-type helical reactor will be reviewed and preferred modeling from the view point of system design will be discussed.

[37] The role of molecular reactions on power, particle and momentum balance during detachment

Presenter: VERHAEGH, Kevin (CCFE)

It has been recognized, using divertor modelling cues, that molecular reactions can play an important role in divertor detachment through additional source/sinks of momentum, particles and energy. Such predictions, however, are difficult to confirm experimentally. We have developed such an experimental method and applied it to TCV discharges. The results support the importance of molecular processes and have implications on the handling of molecules in transport codes like SOLPS-ITER.

The increase of the Dα emission during detachment was quantified and used to infer the role of molecular reactions on power/particle balance during detachment. Based on previous work, higher-n Balmer lines were used to estimate the atomic contributions (recombination and excitation) to Dα. The difference to the total chordally-integrated Dα is used as an estimation of the molecular contributions. These are used to estimate the total hydrogenic radiation from plasma-molecule interactions, the intensity of Molecular Activated Recombination (MAR), and a measure of Molecular Activated Ionisation (MAI). The technique was applied to a TCV discharge, indicating that plasma-molecule interaction with D_2^+ (and/or D^-) can contribute up to 70% of the total Dα divertor emission in detached conditions during density ramps; while no increase beyond the atomic Dα is found during N_2 seeding. As no Dα enhancement is observed during N_2 seeded cases, the D^- and D_2^+ densities are concluded to be significantly lower. Possible reasons for this will be explored in this work.

D_2^+ (and/or D^-), alone, can account for significant radiative losses (up to 40% of all hydrogenic radiation) and significant ion sinks – MAR. The MAR onset occurs after detachment onset (power limitation) but before the onset of electron-ion recombination, (of smaller magnitude than MAR) throughout the discharge. Experimentally, we thus conclude that the influence of plasma-molecule interaction on particle and power balance can be significant. Additionally, SOLPS simulations indicate significant momentum removal can occur from plasma-molecule interaction. The implication of such molecular reactions for ITER/DEMO will be explored.

Analysis of Super-X MAST-U SOLPS simulations has indicated that plasma-molecule interactions, which are strongly enhanced by the large divertor chamber, have strong implications for opacity and its diagnosis. In addition, it is found that vibrational states are crucial in generating D^- and D_2^+ whose densities are strongly underestimated, in the standard use of SOLPS, in comparison to that obtained by post-processing SOLPS output through AMJUEL which matches the measured molecular contribution of Dα during the experiment. There is a need for continued comparisons of experiments with modelling to validate whether the physics involving molecules, as well as their influence on the plasma (particle/momentum/energy sources/sinks), are properly included in SOLPS.

[10] Characterization of liquid metals as prospective divertor materials under transient plasma loads

Presenter: MAKHLAI, Vadym (National Science Center "Kharkov Institute of Physics and Technology", Institute of Plasma Physics)

High power magnetically confined fusion devices have very high heat and particle loads on the plasma facing components. Liquid metals (LM) mock-ups were proposed as alternative of full tungsten divertor for DEMO. Extrapolation of the disruptions/ELMs erosion effects obtained at the present-day tokamaks to the transient peak loads of next step fusion devices (ITER and DEMO) remains uncertain. Special investigations on material behavior at the relevant transient loads are thus very important.

Main features of plasma–surface interaction, vapor shield effects and energy transfer to LM materials are studied at different heat loads within the QSPA Kh-50 [1] and QSPA-M [2]. Repetitive plasma exposures of capillary pore systems (CPS) based Sn targets were performed at the plasma loads varied in the range 0.1–0.5 MJ/m2. Observations of plasma interactions with exposed surfaces were performed with high-speed camera. Optical emission spectroscopy in visible wavelength range in a free plasma stream and within vapor shield layer in a front of exposed LM targets have been measured at different heat loads. The plasma density in this transient layer is found to be up to ten times higher than in impacting plasma stream. It leads to the arisen screening effect for the plasma energy transfer to the surface. Calorimetric measurements of the target heat loads have shown importance of the shielding during the plasma surface interaction. Spectroscopy measurements have shown that for small energy loads the transient layer consists of the plasma stream species only, target impurities are appeared when the heat load exceeds the material melting threshold. The thickness of Sn vapor shield is less than 5 mm. Further increase of the energy load causes the development of strong vapor shield, which dominates in plasma-surface interaction

[1] V.A. Makhlai et. al. Nuclear Materials and Energy 19 (2019) 493–497
[2]. I.E. Garkusha, et. al. 2019 Nucl. Fusion https://doi.org/10.1088/1741-4326/ab1932

[3] Addressing the effect of E×B on closure diveror detachment onset by SOLPS

Presenter: DU, Hailong (Southwestern Institute of Physics)

Addressing the effect of E×B on closure diveror detachment onset by SOLPS

Hailong Du, Guoyao Zheng, Jiaxian Li
Southwestern Institute of Physics, PO Box 432, Chengdu 610041, People's Republic of China
Email: duhl@swip.ac.cn

Abstract

The closed divertor (such as small angle slot-SAS, C-Mod)[1] can well trap neutral (D, D2) and carbon impurity from erosion particles in target region, so that the impurity and recycling neutral particles can radiate a large quantity of power[2]. As a result, the closed divertor very easily achieves detachment with relative small upstream density. However, experimental results in DIII-D and C-Mod [3]showed that the direction of Grad-B has very big influence on the closed diveror detachment onset. As we known, the electric drift- E×B is much related with the direction of Grad-B, and can drive a large number of particles from outer into inner target region along the poloidal flux tubes in private flux region when Grad-B toward divertor. The E×B strongly depends on the radial and polodial gradients of electron temperature-Te and static pressure-neTe[4]. Since the closed divertor can well screen impurity and neutral recycling particles, the polodial gradients of Te and neTe are much larger in closed divertor than that open divertor. Thus, the E×B may have a great effect on the detachment onset in closed divertor. In this work, we will employ the edge plasma code SOLPS to address the effect of E×B on the closed diveror detachment onset in HL-2M. The preliminary modeling results reveal that the effect of E×B on detachment onset in closed divertor is much larger than open divertor, and the closed divertor is very difficult to achieve detachment when Grad-B toward divertor when activating E×B. This work will excite great interests in advancing scientific understanding of the interplay between E×B and impurity energy dissipation for detachment control in closed divertor.

References

[1] Guo H Y, Sang C F, Stangeby P C, Lao L L, Taylor T S and Thomas D M 2017 Small angle slot divertor concept for long pulse advanced tokamaks Nucl. Fusion 57
[2] Sang C, Guo H Y, Stangeby P C, Lao L L and Taylor T S 2017 SOLPS analysis of neutral baffling for the design of a new diverter in DIII-D Nucl. Fusion 57 065043
[3] Lipschultz B, LaBombard B, Terry J L, Boswell C and Hutchinson I H 2007 Divertor physics research on alcator C-Mod Fusion Sci. Technol. 51 369–89
[4] Du H, Sang C, Wang L, Bonnin X, Wang H, Sun J and Wang D 2017 Role of E × B on in–out divertor asymmetry in high recycling/partial detachment regimes under L-mode and H-mode conditions Nucl. Fusion 57 116022

[74] Two-phases hybrid model for neutral gas transport in Soledge2D-EIRENE

Presenter: MARANDET, Yannick (PIIM, CNRS/Aix-Marseille Univ., Marseille, France, EU)

Neutral particles play a key role in addressing the power exhaust challenge in magnetically confined fusion devices. The presence of neutral gas, together with impurities, allows reaching tolerable plasma temperature in front of the divertor targets, and helps reducing peak heat fluxes through power spreading. In fact, the neutral gas pressure in the divertor is often considered to be a key control parameter for divertor conditions. The Knudsen number of the hydrogen isotopes atom gas flow, defined as the ratio of the mean free path to a plasma gradient length, is much larger than one in most of the device. As a result, neutrals have generally been treated kinetically for the last 35 years. In view of the geometrical complexity, of the numerous species and reaction channels that have to be taken into account, the available solvers rely on a Monte Carlo approach. In next generation devices, simulations show the formation of high collisionality regions of limited spatial extent in the divertor, where particles tracked by the Monte Carlo code can get effectively trapped, undergoing a large number of collisions before being ionized or escaping. This has the effect of considerably slowing down simulations and can be a major hurdle for scoping/design studies. In these high collisionality regions, the neutral gas is expected to behave like a fluid, with a Maxwellian distribution, and using a hybrid kinetic/fluid model is tempting, as done in other applications (e.g. atmosphere re-entry). Two main classes of models have been developed in the literature, namely micro/macro approaches and domain decomposition models. The former solves for kinetic corrections over the whole domain and the later introduces boundaries between kinetic and fluid regions. In practice both approaches introduce modelling errors. The approach based on domain decomposition requires setting up criteria to decide whether a region is kinetic or fluid. In this contribution, we show that the two-phase hybrid model proposed in Ref. [1] can be understood as a particular type of domain decomposition approach. The model has been implemented for atoms in Soledge2d-EIRENE [2], and essentially consists in introducing artificial reaction channels, with condensation terminating trajectories entering into highly collisional regions. EIRENE thus calculates a source of fluid atoms, which is transferred to a neutral fluid code. In low collisionality regions, fluid atoms evaporate into kinetic atoms. The latter are treated as a volumetric source in the neutral fluid code. Here we take a closer look at the sensitivity of the solution obtained in pure deuterium ITER cases on the model parameters: i) form of the criteria defining fluid regions ii) sharpness of the transition layer, ii) amplitude of the condensation/evaporation rates. Modelling errors are assessed by comparison to pure kinetic calculations, and the computational gain is estimated for each case. These gains would ease the contribution of plasma edge modelling to the divertor design process.

[1] C. F. F. Karney, D. P. Stotler and B. J. Braams, Contrib. Plasma Phys. 38, 319 (1998).
[2] M. Valentinuzzi et al., Nuclear Materials and Energy 18, 41 (2019).

[41] Liquid Metal Modeling for Plasma Facing Components

Presenter: KHODAK, Andrei (Princeton Plasma Physics Laboratory)

Liquid metal (LM) plasma facing components (PFC) are considered an attractive design choice for fusion devices including pilot plants. Several liquid metal concepts for the divertor region are currently under development. Lithium or lithium eutectics have a high affinity for tritium and deuterium at low operating temperatures, and provide a low-recycling boundary condition for the core plasma, which can lead to significant confinement improvements. Liquid metals have sufficient thermal conductivity to control their temperature below evaporation point. However LM can also provide vapor shielding of PFC under transient heat loads. On the other hand, electrically conducting liquids are subject to magnetohydrodynamic (MHD) interactions that can disturb the LM and eject its material into the plasma. MHD effects can also laminarize the flow, thereby reducing turbulence and convective heat transport. In addition, MHD drag can impose additional requirements on the LM delivery system across the magnetic field of the device. An important characteristic of fusion relevant liquid metal flow is free surface smoothness and stability. Heat flux from the plasma impacts the liquid surface at a grazing angle, therefore any change of the free surface conditions can dramatically increase the local heat flux density and therefore create excessive evaportion of liquid lithium into the plasma.

Solid metallic PFCs can also undergo transient melting in response to high heat flux events such as large edge-localized modes (ELM), unipolar arcs and disruptions. Changes in surface morphology caused by the motion and possible destabilization of the resulting melt layer can lead to a considerable degradation of the PFC longevity and heat-handling properties.

Virtual prototyping has proved useful to address some of the issues outlined above. Modeling of free-surface flows of electrically conductive liquids is facilitated by computational fluid dynamics (CFD) and MHD simulations. Moreover, phase transitions can alter the heat balance and, depending on the time scales involved, resolidification and evaporation can also significantly affect the dynamics of the liquid. Coupling fluid dynamics and MHD solvers to heat transfer enables one to adress such problem fully self-consistently. LM film flow of different thicknesses was analysed.

Numerical tools capable of simulating flows and heat transfer in the free-surface MHD flow were developed at PPPL based on the customized ANSYS code. MHD is introduced using a magnetic vector potential approach. Free-surface flow capabilities are available in the code and were tested. Electro-magnetic equations are solved in the liquid metal, as well as in the solid components of the structure and plasma. Special stabilization procedures were derived and applied to improve convergence of the momentum equations with the source terms due to the Lorentz force and surface tension.

The same set of numerical tools was successfully adopted at PPPL for modeling atmospheric pressure arcs and was thoroughly validated by comparison with experimental data. Recently we adapted the model to simulate transient metal melting and splashing under conditions corresponding to unipolar arcs on tokamak PFCs. The numerical solutions accounting for phase transitions, are benchmarked against available simulation results and experimental data.

[61] Results of a model of interchange turbulent transport on the correlation between scrape off layer width and core confinement in tokamaks

Presenter: FEDORCZAK, Nicolas (CEA)

Turbulent transport has two critical impacts on the operational domain of tokamak reactors: it sets the core confinement performances through limitation of kinetic gradients from the very centre of the confined plasma to the magnetic separatrix, and it sets the condition of power exhaust by the tokamak wall, through the size of the heat flux wetted area. Experiments across a variety of magnetic configurations tell us that improvement of confinement is correlated with a reduction of the heat flux wetted area, as clearly established by the comparison of low and high confinement mode experiments. In that respect, research on advanced divertor concepts has to find a way to disentangle power exhaust conditions from that of confinement, for instance by optimizing both turbulent spreading of the wetted area and dissipation performances of the boundary plasma volume. A key aspect of this research is to understand the transport mechanisms taking place from the core to the plasma periphery, and how they correlate.

A model of transport by interchange turbulence was recently derived to propose a physical basis of the experimental scaling on scrape off layer width. The model is found to accurately predict global and local properties of scrape off layers in circular limited geometries: width of the wetted area, fluctuation levels of density and electrostatic potential, scale of the dominant turbulent modes, etc. Even though the model overpredicts the scrape of layer width of divertor configurations, suggesting missing physical ingredients from a magnetic X-point, sensitivity with global control parameters is well recovered. In addition, the model is able to recover recent experimental findings from the TCV tokamak where the width of the wetted area was found to increase significantly with the geometrical length of the outer divertor leg. These results suggest that the model contains the ruling physics of scrape off layer transport even in diverted geometry. Another striking result from the model is the capability to predict a global energy confinement time that is accurately matching the trend of the multimachine scaling, apart from a multiplication factor. As found in experiments, the model therefore predicts a close correlation between global confinement performances and power exhaust conditions, principally driven by the plasma current. But it also offers an interesting physical basis to optimize power exhaust at least through geometrical considerations: Can we expect a given advanced divertor geometry to efficiently enhance interchange spreading of the heat flux wetted area?

[6] Assessment of vapor shielding efficiency in lithium divertor for steady-state and transient events

Presenter: MARENKOV, Evgeny (National Research Nuclear University MEPhI, Moscow, Russian Federation)

Large heat loads on the divertor targets, especially during transient events are one of the problems envisaged for industrial thermonuclear reactors. Liquid renewable coatings are considered as a possible alternative to now accepted full-metal divertor design. Along with protecting the underlying divertor and first wall materials from direct plasma exposure, lithium (Li) as a plasma facing material has also positive influence on the main plasma performance, e.g. enhanced confinement, ELM suppression [1,2].

Estimation of power loads on divertor targets covered by Li has to include vapor shielding effects. Corresponding calculations are conducted using both sophisticated edge plasma codes [3] and very simple analytical models [4]. However, these studies are focused on the steady-state regimes, whereas Type-I ELMs and disruptions are more threatening to the plasma facing components. Therefore, it is reasonable to assess the Li power loads mitigated by shielding effects in non-steady-state situation.

In this work we present a 0D model suitable for such estimations. The model includes Li erosion both due to sputtering and evaporation, surface recombination of plasma and Li ions, elastic and charge-exchange collisions between all species in the vaporized cloud, volumetric ionization and recombination, and radiation. The surface temperature is calculated from a 1D thermo-conductivity equation.

Using the model, we consider steady-state shielding and the coating behavior under ELM-like heat pulses. Steady-state calculations show that Li coating can sustain heat loads larger than 10 MW/m^2 if its renewal rate is high enough, $10^{25} \mathrm{m^{-2}s^{-1}}$. These results do not contradict previous estimates based on ‘energy lost per evaporated particle’ approach [4].

Authors of [4] have also considered impact of pulsed heat-loads on a Li target. However, the estimates suppose that the energy lost per evaporated particle does not change during pulses. This approximation does not take into account neither finite time of the vapor cloud formation, or evolution of the plasma parameters, which influence the ‘energy lost per evaporated particle’. Our calculations are free of these deficiencies. We consider the vapor shielding under pulsed heat loads typical for type-I ELMs. We show that shielding on average significantly reduces heat loads on the target. Necessary renewal rates for the Li coating to survive in such regimes are estimated.

This work was supported by the Russian Science Foundation grant 18-12-00329

References

[1] S. Mirnov, J. Nucl. Mater. 390–391 (2009) 876–885.
[2] M. Ono et al., Nucl. Fusion 53 (2013) 113030.
[3] T.D. Rognlien et al., Nucl. Mater. Energy 18 (2019) 233–238.
[4] P. Rindt et al., Nucl. Fusion 58 (2018) 104002.

Wednesday 06 November 2019

Registration - Board Room C (C Building, 4th Floor) (6 Nov 2019, 08:30-09:00)

Arrival of participants, distribution of badges, possibility to register and pay for events.

Core-Boundary Plasma Compatibility - Board Room C (C Building, 4th Floor) (6 Nov 2019, 09:00-10:20)

time [id] title

09:00 **[81] Core-pedestal constraints on divertor design**

Presenter: LEONARD, Anthony W. (USA)

For reactor-scale tokamaks, the core plasma operational scenario imposes a number of boundary conditions on the divertor and SOL plasma. The most critical of these, upstream power density flowing into the divertor, and upstream separatrix electron and impurity density exert high leverage over divertor designs, and may even determine the need for advanced divertor designs to safely exhaust the power and particle flux from the core plasma. This presentation will summarize recent work to develop the physics basis for projecting these upstream boundary conditions to future DEMO scale fusion plasmas. For heat flux density flowing into the divertor for existing devices an empirical scaling of the heat flux width consistent with plasma drifts has proved valuable for predicting heat flux in existing devices. However, turbulence codes such as XGC and BOUT++ suggest that in future devices with larger ratios of size to ion gyro-radius, a/rI, higher levels of turbulent transport would increase the heat flux width and thereby relax the requirement for divertor heat flux dissipation. Edge MHD instabilities may also play a role in the level of heat flux dissipation that future divertors may face. The lower level of central fueling in DEMO-scale devices compared to edge recycling in existing devices could lead to flatter pedestal density profiles with significant consequences for divertor design. A flat pedestal density profile, such as that predicted for ITER, would allow for larger values of upstream separatrix electron density for a given core scenario density and allow for greater divertor heat flux dissipation. A flat pedestal density would also decrease the pedestal inward neoclassical pinch that causes core impurity accumulation in H-mode in existing devices. This would allow for greater low-Z impurity seeding for higher levels of divertor dissipation. Recent experimental efforts to study pedestal density transport include cross-machine scalings and variations of divertor geometry to examine the response of the pedestal density profile to changes in fueling. Complementary to the experimental studies are emerging numerical models capable of simulating plasma turbulence and transport in the steep gradient regions of the pedestal. The radial fluxes predicted by the models for measured pedestal profiles are now being compared to those inferred from experiment. Further issues and ongoing work to describe the pedestal-SOL interface will be discussed.

09:30 **[15] Role of transport versus fueling upon the pedestal density**

Presenter: MORDIJCK, Saskia (The College of William and Mary)

Fueling a future fusion reactor and the effects of fueling upon the pedestal structure is an open physics question and the U.S fusion research program has initiated a multi-institutional effort to develop a physics basis to address this question. In contrast to existing devices, future reactor-scale tokamaks the edge pedestal will be opaque to neutrals and no longer be fueled primarily by edge recycling. As both non-normalized plasma density and machine size increase, we reduce the ability of recycling or gas puff neutrals to penetrate the confined plasma and to contribute to particle sources on closed flux surfaces. We can parameterize the screening of fueling neutrals according to roughly $\delta \sim n \times a$, based on the boundary region width and this is often taken as a rule-of-thumb neutral opacity.

Recent experiments on C-Mod ($n_e \times a~3x10^{20} \times 0.22 $) and DIII-D ($n_e \times a~0.9x10^{20} \times 0.67$) to investigate the role of neutral opacity go from 10 less opaque then ITER, to only a factor 2 less opaque then ITER predictions. In these experiments we observe that the density pedestal structure has moved into the SOL region for C-Mod, but that counter to prior JET and AUG results, this does not result in a degradation of the pedestal pressure. Moreover, in DIII-D, large gas puffs (up to 300 TorrL/s) to increase the opacity, resulted in an increase in the pedestal pressure up to 200 TorrL/s and degradation was only observed at 300 TorrL/s. SOLPS-ITER modeling of these DIII-D and C-Mod discharges show a decrease in neutral penetration with increased opacity and a change in the poloidal distribution of the neutrals, with a deeper penetration from the main vessel on the LFS at high opacity, versus a more uniform distribution at lower opacity.

In regular, DIII-D H-mode discharges (half the opacity of those reported above), the divertor geometry affects the density pedestal profile. A closed divertor geometry shows a reduced density gradient consistent with estimates of a reduction in pedestal fueling. This is a clear indication that changes to pedestal fueling can significantly affect the edge pedestal density structure. The implications of these results towards scaling to future tokamaks with opaque SOLs will be discussed.

While we observed a density pedestal structure in C-Mod, at the highest opacity which resulted in the highest plasma pressure, fueling through a gas puff is very inefficient. Recent DIII-D experiments and NSTX experiments show that for the same fueling rate, pellets and supersonic gas injections are much more efficient, due to deeper penetration of the fuel.

Complementary to the experimental observations, numerical models such as GENE and TGYRO, are capable of simulating plasma turbulence and transport in the steep gradient regions of the pedestal The radial fluxes predicted by the models for measured pedestal profiles are now being compared to those inferred from experiment.

Work supported by US DOE under DE-FC02-04ER54698, DE-AC02-09CH11466, DE-SC0014264, DE-SC0019302, DE-SC0007880

10:00 **[47] Radiative Power Exhaust Research at DIII-D - From Divertor Science to Core-Edge Integration of High Performance Plasmas**

Presenter: JAERVINEN, Aaro (Lawrence Livermore National Laboratory)

Power exhaust research at DIII-D is addressing the need for fusion reactors to integrate high β_N, high confinement plasmas with a radiating mantle and divertor that are compatible with stringent requirements on fuel dilution, high confinement mode power threshold, plasma stability, and wall heat flux limits. This research program encompasses studies from diagnostic optimized divertor science to compatibility of power exhaust with high performance pedestal and core plasmas [1, 2, 3].

Interaction of radiative divertor with high performance, peeling limited pedestals have been investigated in the super H-mode regime in DIII-D, reaching 2$p_\text{e,ped}$ > 20 kPa. By using a highly shaped configuration with a particular density trajectory, a peeling limited pedestal can be maintained at density levels where ballooning modes would typically reduce the achievable $p_\text{e,ped}$. Using 3D coils to control $n_\text{e,ped}$ while increasing gas fueling, $n_\text{e,sep}$ close to 50% of $n_\text{e,ped}$ were achieved without compromising $p_\text{e,ped}$. With N$_2$ injection, the radiative power fraction, f_RAD, was increased from 40% to 65%, providing a factor of 3 reduction of the outer target electron temperature while $p_\text{e,ped}$ was maintained. However, increasing f_RAD to 85% was observed to lead to strong divertor detachment with a highly localized radiation at the X-point, and about a factor 2 reduction in $p_\text{e,ped}$.

In hybrid plasmas at high β_N > 3.0, divertor peak heat fluxes could be reduced by about 40% with either neon or argon injection. However, fuel dilution and the emergence of harmful tearing modes significantly compromised the performance of these plasmas, in contrast to previous observations at lower β_N operation. These findings highlight the need to control impurity transport and tearing mode stability when integrating mantle radiation to a DEMO relevant plasma core.

To test the scaling of impurity densities required for divertor detachment and its compatibility with a high performance core plasma, diagnostic capability to directly measure divertor radiating impurity densities has been developed on DIII-D. By combining divertor Thomson scattering with vacuum ultraviolet spectroscopy, the local emitting ion density can be measured directly.

Work supported by the US DOE under DE-FC02-04ER54698 and DE-AC52-07NA27344,
and LLNL LDRD project 17-ERD-020.

[1] A. E. Jaervinen, et al. Nucl. Mat. Ene. 19 (2019) 230-238
[2] P. B. Snyder, et al. Accepted to Nucl. Fusion 2019
[3] T. W. Petrie, et al. Submitted to Nucl. Fusion 2019

Coffee Break - Board Room C (C Building - 4th Floor) (10:20-10:40)

Core-Boundary Plasma Compatibility: Continued - Board Room C (C Building, 4th Floor) (6 Nov 2019, 10:40-11:40)

time	[id] title
10:40	**[43] Experimental studies of the nitrogen concentration required for divertor detachment in ASDEX Upgrade** *Presenter: HENDERSON, Stuart (UKAEA)* Substantial seeding of impurities into the divertor volume has been used for a long time in tokamaks to reduce the power and particle fluxes impacting on the divertor targets and is one of the main techniques to be utilised on ITER to allow safe, steady state divertor operation. Since the amount of power radiated by the impurity species varies directly with the electron and impurity ion density, the impurity concentration in the scrape-off layer (SOL) is therefore a necessary input for predictive scaling of divertor detachment in future devices. There has been a recent effort to develop scaling laws [1-3] to predict the impurity concentration required for detachment and to assess how these concentrations vary with different plasma parameters, such as the power crossing the separatrix, Psep, the separatrix density, ne,sep, and the poloidal magnetic field. However, due to the difficulty of the measurement itself, currently there are no experimental studies to validate these predictions and guide expectations for ITER and DEMO. To try and address this, the work presented here builds on previous preliminary measurements of the nitrogen concentration in the divertor volume, cN, determined from chord-averaged N II line emission [4,5] and examines the specific parameter dependencies and trends with respect to the detachment state. From a database of ASDEX Upgrade N-seeded H-mode discharges, spanning Psep=4 – 11 MW and ne,sep=1.5 – 3x10^19 m-3, with total injected powers ranging from 6 – 14 MW, line averaged core densities from 7 – 8x10^19 m-3, and plasma currents from 0.8 – 1 MA, the cN measurements at the onset of detachment will be presented and shown as a function of Psep, ne,sep, and scaling law calculations. The measurements suggest that the concentrations scale approximately linearly with Psep and inversely with the square of ne,sep; however the absolute values are approximately four times lower than scaling law predictions. A detailed comparison of the plasma scenarios chosen in this database will be presented, with a focus on the strike-point position along the outer divertor target plate and how this effects the length of the N II emission volume along the line-of-sight. The time window relative to the detachment state through which the cN is averaged will be assessed, providing data before and during divertor detachment. Finally, the relevance of the measurement in the divertor compared to the average SOL quantities used in scaling predictions will be discussed. [1] A. Kallenbach et al. 2016 Plasma Phys. Control. Fusion 58 045013 [2] R. Goldston et al. 2017 Plasma Phys. Control. Fusion 59 055015 [3] M. Reinke 2017 Nucl. Fusion 57 034004 [4] S. Henderson et al. 2018 Nucl. Fusion 58 016047 [5] S. Henderson et al. 2019 NME 18 147-152
11:00	**[66] Progress towards understanding ITER's divertor heat-flux width from gyrokinetic simulation** *Presenter: CHANG, C.S. (Princeton Plasma Physics Laboratory)* While prediction for the divertor heat-flux width in the wide range of the poloidal magnetic field in attached NSTX, DIII-D, NSTX and JET plasmas matches the Eich-14 scaling formula within the regression error bar, the XGC prediction for the full-current ITER showed over six times wider heat-flux width than the Eich value. There were questions from the community if this difference in the simulation results is from the large size effect, small ion gyroradius effect, or the large a/ρi effect (neoclassical), where a is the plasma minor radius and ρi is the ion gyroradius. More simulations of a lower current ITER H-mode and the recent smallest-ρi C-Mod plasmas again show agreement with the Eich formula, rejecting the possibility for the large size or small ρi effect. Deeper analyses of the simulation data, including a machine learning study, show evidence for different edge physics (trapped electron modes instead of blobs) being in play in ITER edge that spreads the divertor heat-flux wider. When the a/ρi effect is added to the Eich formula, distance between the smallest-ρi C-Mod and the full-current becomes very far in the parameter space removing the suspicion of an abrupt bifurcation in the ρi space. Study of the NSTX-U plasma further supports the spreading action of the heat-load by trapped-electron-modes. Most up-to-date understanding of the underlying physics and suggestions for corresponding experimental explorations on today's tokamaks will be discussed. *Funding provided by US DOE Office of Science FES and ASCR. Computational resources provided by OLCF, ALCF and NERSC

11:20 **[16] On the Divertor Heat Flux Width Scaling**

Presenter: XU, Xueqiao (Lawrence Livermore National Laboratory)

The BOUT++ code has been used to simulate edge plasma electromagnetic (EM) turbulence and transport, and to study the role of EM turbulence in setting the scrape-off layer (SOL) heat flux width λ_q. More than a dozen tokamak discharges from C-Mod, DIII-D, EAST, ITER and CFETR have been simulated with encouraging success. The parallel electron heat fluxes onto the target from the BOUT++ simulations of C-Mod, DIII-D, and EAST follow the experimental heat flux width scaling of the inverse dependence on the poloidal magnetic field. Further turbulence statistics analysis shows that the blobs are generated near the pedestal pressure peak gradient region inside the separatrix and contribute to the transport of the particle and heat in the SOL region. Transport simulations show two distinct regimes: drift dominant regime and turbulence dominant regime. For current tokamak H-mode discharges, drift and turbulent transport both compete in setting the heat flux width, possibly due to its compact machine size and good pedestal confinement.

The simulations for ITER and CFETR indicate that divertor heat flux width □q of the future machines may no longer follows the 1/Bpol,MP experimental empirical scaling and the Goldston HD model gives a pessimistic limit of divertor heat flux width. The simulation results show a transition from a drift dominant regime to a turbulence dominant regime from current machines to future machines such as ITER and CFETR for two reasons. (1) The magnetic drift-based radial transport decreases due to large CFETR and ITER machine sizes. (2) the SOL turbulence thermal diffusivity increases due to larger turbulent fluxes ejected from the pedestal into the SOL when operating in a different pedestal structure, from an ELM-free H-mode pedestal regime to a small/grassy ELM regime.

Experimental observations on C-MOD and ASDEX-Upgrade have shown a similar parametric dependence for λ_q in I-mode and L-mode discharges to that previously found in H-mode, with consistently larger width and weaker current I_p^(-α) scaling as turbulence transport increases. But turbulence seems not strong enough to dominate the radial transport by drifts in I-mode and L-mode discharges, and hence to break the experimental scaling (Eich scaling), in which λ_q□1/I_p (or 1/Bpol). Further researches show that grassy ELM operation yields larger turbulence transport with H-mode level performance across a range of collisionalities. A stationary, high-confinement small grassy-ELM regime has been demonstrated in EAST with fELM~2.6kHz and q┴<2MW/m2 on divertor target, which is less than 10% of that of type-I ELM. Furthermore, recent DIII-D grassy ELM experiments show a consistent divertor heat flux width broadening and amplitude reduction, just as BOUT++ simulations demonstrated in the grassy ELM regime. The comparison of divertor heat flux width will be given for quasi-steady inter-ELMs H-mode, grassy ELM and type-I ELM discharges.

Discussion Session: Core-Edge - Board Room C (C Building, 4th Floor) (6 Nov 2019, 11:40-12:30)

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

DEMO Divertor Designs - Board Room C (C Building, 4th Floor) (6 Nov 2019, 12:30-13:20)

Plenary sessions collect all contributions invited or accepted by the Programme Committee for a complete in depth plenary session.

time	[id] title
12:30	**[19] Overview of the physics and diagnostics modelling activities for the EU-DEMO divertor** *Presenter: SICCINIO, Mattia (EUROfusion Consortium)* The current baseline EU-DEMO, as designed by the EUROfusion Power Plant Physics & Technology Department (PPPT), considers an ITER-like LSN divertor for particle and power exhaust. Various modelling activities have been undertaken in the past years to assess the performance of this key machine component, both concerning plasma physics and engineering design. Goal of the present work is to provide an overview of the status of the ongoing investigations related to plasma physics and diagnostics (concerning the steady-state phase and the off-normal transients, but disregarding the role of ELMs). The EU-DEMO divertor is designed to withstand a steady-state heat flux of 10 MW/m2 and higher loads during transient events, while implementing a considerable safety margin to account for uncertainties. At the same time, the temperature of the impacting plasma shall be below 5 eV to avoid excessive sputtering of the W armour. These requirements can only be met when the divertor plasma is in a detached state. For detachment, it is necessary to dissipate a large fraction of the power crossing the separatrix using seeded radiative impurities. Both fluid and kinetic SOLPS simulations have been performed to determine the conditions under which a robust detachment can be obtained and maintained, with particular attention to the evaluation of the necessary seeded impurity fluxes. Also, SOLPS results have been employed as boundary conditions for the codes DIVGAS and ITERVAC, to assess the He removal from the plasma chamber by the pumping systems. Such removal shall in fact occur at a rate equal to He generation from the fusion reactions, in order not to let the concentration in the core increase up to a level where the reactor performance deteriorates. A further aspect of particular importance in the EU-DEMO design is the protection of the divertor for a time sufficient to ramp-down the plasma current in case detachment is lost. This is necessary as the fragility of the breeding blanket wall does not allow for a loss of plasma control at full current following a fast shutdown. Currently, strike point sweeping by means of in-vessel magnetic coils is the favoured option. This strategy assumes the existence of divertor detachment diagnostics that allow the detection of reattachment with sufficient time resolution. Spectroscopy and thermocurrent measurements, foreseen to perform this task, are presented and discussed. Finally, investigations performed by coupling the transport code ASTRA to a 0-D SOL and divertor model, with the purpose of evaluating the impact on the core plasma of the SOL seeded impurities, are illustrated. Results concern both steady-state simulations as well as dynamic cases, modelled with the ASTRA/Simulink DEMO control simulator.

13:00 **[79] Insights from Systems Code Analysis on Power Exhaust Requirements for Future Fusion Power Systems**

Presenter: WADE, Mickey R. (General Atomics)

Recent analysis using the GA systems code (GASC) has provided new insights into the power exhaust requirements for future fusion power systems. This analysis was enabled by improvements in the underlying models for power exhaust, magnet technology limits, bootstrap current, and costing as well an improved optimization algorithm capable of identifying optimum solutions for a range of assumptions and constraints. This analysis has confirmed the importance of power exhaust limitations in systems with assumed confinement quality consistent with the confinement scaling used for ITER (H98y2 = 1.0). Under such an assumption, the device size and associated cost is critically dependent on the assumed heat flux handling capability of the facility. For example, decreasing the maximum heat flux assumption from 10 MW/m2 to 5 MW/m2 for a 200 MW-e pilot plant results in an increase in device size from 6.75 m to 8.4 m (and a 20% increase in estimated capital cost). At this level of confinement, the power exhaust limitations serve as the primary constraint on the plasma device size and associated cost.

Importantly, GASC analysis indicates that this sensitivity to the assumed heat exhaust limits decreases strongly as confinement quality increases even as the overall size of the device decreases significantly. In fact, at high confinement quality (e.g., H98y2 = 1.7), the sensitivity largely disappears with other factors such as the stability limit and neutron wall loading becoming the dominant factors. Detailed analysis suggests that there are two contributing factors to this reduced sensitivity. First, as H98y2 increases, the overall heating requirement to maintain the necessary plasma energy (and hence fusion power producing capability) increases markedly. Second, higher H98y2 reduces the required level of plasma current required to provide the necessary level of absolute confinement. The reduced plasma current leads to an expansion of the SOL heat flux width and a concomitant decrease in the divertor heat flux. These effects are sufficient to overcome the impact of the smaller device size on the anticipated heat flux.

The improved heat flux models now employed GASC have also allowed exploration of the value of various techniques for reducing the heat flux. These include the ability to systematically evaluate the best impurity (or impurities) for core and divertor radiation as well as the potential gains possible through poloidal flux expansion, toroidal flux expansion, and target inclination. This along with further analysis will be presented.

This work was supported by General Atomics corporate funding.

Lunch - Board Room C (C Building - 4th Floor) (13:20-14:30)

DEMO Divertor Designs: Continued - Board Room C (C Building, 4th Floor) (6 Nov 2019, 14:30-15:10)

Plenary sessions collect all contributions invited or accepted by the Programme Committee for a complete in depth plenary session.

time [id] title

14:30 **[25] Investigation of divertor operation for Japanese DEMO under low density SOL and large power exhaust of P_{sep}/R ~30 MW/m level**

Presenter: ASAKURA, Nobuyuki (National Institutes for Quantum and Radiological Science and Technology (QST), Naka Fusion Institute)

$~~$ Power exhaust scenario for the feasible DEMO plasmas and the divertor design have been studied with a high priority in the steady-state Japanese (JA) DEMO with the fusion power of 1.5 GW-level and the major radius of 8 m-class. The power exhaust concept requires large power handling in the SOL and divertor, i.e. P_{sep}~250 MW, and P_{sep}/R_{p}~30 $\rm MWm^{-1}$ corresponds to 1.8 times larger than ITER. Moreover, the SOL plasma density (n_{e}^{sep}) is expected to be relatively low, i.e. $\rm 2-3x10^{19} m^{-3}$, which corresponds to 1/3-1/2 of the pedestal density (n^{ped} ~$\rm 6x10^{19} m^{-3}$) since n^{ped} is restricted by the Greenwald density limit (n^{GW} = $\rm 6.6x10^{19} m^{-3}$). The long leg divertor (L_{div} = 1.6 m; 1.6 times longer than ITER) was proposed as a reference design. SONIC simulation with Ar impurity seeding demonstrated the peak heat load (q_{target}) on the outer target was reduced to less than 10 $\rm MWm^{-2}$ under the partially detached condition with the large radiation fraction of $f_{rad}^{sol+div}$ = ($P_{rad}^{sol}+P_{rad}^{div})/P_{sep}$ ~0.8.

$~~$ Recently, operation of the plasma detachment and acceptable q_{target} ($\le$10 $\rm MWm^{-2}$) has been investigated under the severe conditions such as higher P_{sep} ~300 MW or lower $f_{rad}^{sol+div}$ ~0.7. The peak q_{target} was generally reduced with increasing n_{e}^{sep}, because the detached plasma width was decreased and the peak T_{e}^{div} and T_{i}^{div} at the attached plasma region were increased. The former case (higher P_{sep}) is acceptable in the expecting n_{e}^{sep} range of $\rm 2-3x10^{19} m^{-3}$, while the margin of the peak q_{target} is reduced. On the other hand, the latter case (lower $f_{rad}^{sol+div}$) requires operation in higher n_{e}^{sep} $\rm >2.3x10^{19} m^{-3}$. At the same time, the peak T_{e}^{div} and T_{i}^{div} were increased, which would enhance net erosion of the target. Investigation of the plasma diffusivity also started. Simulations with reducing both χ and D to half values, i.e. χ_{e} =χ_{i} = 0.5 $\rm m^{2}s^{-1}$, D = 0.15 $\rm m^{2}s^{-1}$, were performed for the three cases. Decay length of the parallel heat flux profile near the outer midplane separatrix ($\lambda_{q//}^{sol-OM}$) was decreased from the reference case (~1.9 mm), which is already narrow compared to 3.6 mm in the ITER simulation [2]. Peak q_{target} values were increased for all reduced χ and D cases, and the reference and higher P_{sep} cases were still acceptable, whereas higher n_{e}^{sep} was preferred. On the other hand, for the lower $f_{rad}^{sol+div}$ case, peak q_{target} $\le$10 $\rm MWm^{-2}$ was not achieved in the operation range of the low n_{e}^{sep}. Changes in the characteristics of the q_{target}, T_{e}^{div} and T_{i}^{div} profiles in the partially detached divertor are summarized.

$~~$ In addition, power exhaust scenario with impurity seeding other than Ar will be compared to the reference scenario. Effects of the divertor geometry (angles of target and reflectors, dome height) on the detached plasma will be discussed for the future design optimization with the particle exhaust.

[1] N. Asakura, et al., Nucl. Fusion 57 (2017) 126050.
[2] A. Kukushkin, et al., J. Nucl. Mater. 438 (2013) S203.

14:50 **[40] Assessing Alternative Divertors for DEMO – strategy and first results**

Presenter: MILITELLO, Fulvio (Culham Centre for Fusion Energy)

The uncertainties surrounding the physics of plasma exhaust and its centrality in reactor design require a thorough evaluation of promising exhaust configurations, so EUROfusion established a project to assess alternative divertors for reactor relevant devices and DEMO in particular. An alternative here is any divertor solution that cannot be qualified by ITER and it includes, but is not limited to, Double Null, Snowflake, Super-X and X divertors.

From 2019 the project has a revised strategy and it is delivering new systematic results, presented here for the first time. It focuses on a clear and quantitative evaluation of the potential benefits that alternatives might have with respect to conventional solutions as far as exhaust performances and core compatibility are concerned. These will be weighed against their unavoidable additional engineering complexity, which might rule out some solutions. Additionally, it will drive new ideas in the form of optimized, hybrid or novel solutions, noting the deadline to complete the assessments.

The approach is to deliver integrated results through a “main loop” where physics and engineering concepts are synergistically iterated and optimized. In parallel more refined techniques and new concepts are developed that will become part of the “main loop” when (or if) they reach maturity.

The results of the first iteration of the “main loop” show that all alternatives investigated (Double Null, Super-X, X-divertor and Snowflake divertor) could be generated with only external coils while respecting engineering force constraints on the poloidal field coils and central solenoid. New detailed structural calculations of stresses in the toroidal field coils show that some configurations are subject to much larger stresses. These are mostly caused by out of plane forces, possibly needing more rigid supports, so unconventional toroidal field coil designs may still be acceptable, although not all are equally efficient. Full 3D builds, including intercoil structures and port access for remote maintenance have been produced and their technical complexities will be discussed. For each configuration, vertical stability was also taken into account. On the physics side, multi-fluid scans of core and wall fueling and seeded argon concentration were performed to assess the detachment quality at an unprecedented level by comparing each design for an identical and large range of fueling and argon fluxes. The analysis is now performed with state of the art tools (SOLPS) to enable good understanding of complex effects and prepare for full kinetic simulations. Also, the resilience of the operating point to variations is systematically examined, leading to a full map of the plasma response to changes in fueling and seeding. Hence, detachment onset, depth, stability are obtained, rather than just the operating point. Also, pumping efficiency was assessed. As preliminary conclusions, snowflake configurations appear to be challenging from a structural and remote handling point of view. The other configurations do not present showstoppers, although more complex than the baseline single null design. The Super-X configuration seems to provide more heat flux mitigation benefits, but it might complicate the power handling at the inner target, still under investigation.

Discussion Session: DEMO Design - Board Room C (C Building, 4th Floor) (6 Nov 2019, 15:10-16:10)

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

Coffee Break - Board Room C (C Building - 4th Floor) (16:10-16:30)

Poster Session III - Board Room C (C Building - 4th Floor) (6 Nov 2019, 16:30-18:30)

All submissions accepted as "Poster" should provide a poster conforming to the rules published in the meeting announcement. Posters will be shown outside of Board Room C near by the coffee area.
Board numbers correspond to Indico [ID] numbers.

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[24] Engineering integration constraints for advanced magnetic divertor configurations in DEMO

Presenter: KEMBLETON, Richard (EUROfusion, CCFE)

The divertor configuration defines the power exhaust capabilities of DEMO as one of the major key design parameters and sets a number of requirements on the tokamak layout, including port sizes, PF coil positions, and size of TF coils. It also requires a corresponding configuration of plasma-facing components and a remote handling scheme to be able to handle the cassettes and associated in-vessel components the configuration requires.

Alternative magnetic configurations to that baseline ITER-like single-null (SN) – double-null, snowflake, X-, and super-X – exist and potentially offer power-handling solutions to the limits imposed by plasma-facing component technology and first-wall protection whilst maintaining good core plasma performance. But these options impose significant changes on machine architecture, increase the machine complexity and affect remote handling and plasma physics and so an integrated approach must be taken to assessing the feasibility of these options.

In this contribution we describe the engineering and physics limitations which must be respected in assessing the impact of incorporating these alternative configurations into DEMO, including requirements on remote handling access, forces on coils, plasma control and performance, etc.

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[78] Progress of divertor design concept for Japanese DEMO

Presenter: ASAKURA, Nobuyuki (National Institutes for Quantum and Radiological Science and Technology (QST))

Power exhaust scenario for the feasible DEMO plasmas and the divertor design have been studied with a high priority in the steady-state Japanese (JA) DEMO with the fusion power of 1.5 GW-level and the major radius of 8 m-class. The power exhaust concept requires large power handling in the SOL and divertor, i.e. Psep~250 MW, and Psep/Rp~30 MWm-1 corresponds to 1.8 times larger than ITER. Long leg divertor (Ldiv = 1.6 m; 1.6 times longer than ITER) was proposed as a reference design. SONIC simulation demonstrated that the peak heat load on the target (qtarget) was reduced to less than 10 MWm-2 under the partially detachment with large radiation fraction of (Pradsol+Praddiv)/Psep ~0.8. A design concept of the monoblock target and cooling water pipes for the JA DEMO was proposed in 2016 [1]: two different water-cooling units, i.e. 200□C, 4MPa pressurized water in CuCrZr pipe and 290□C, 15MPa pressurized water in Reduced Activation Ferritic–Martensitic steel (F82H) pipe, are used. The heat exhaust unit with the CuCrZr pipe can be applied near the strike-point (0.8 m) for the high heat load region, while the replacement is expected every 1-2 years due to the maximal irradiation dose on the CuCrZr pipe of 2 displacement-per-atom (DPA).
Recently, cassette structure for the DEMO divertor was designed to incorporate the heat exhaust units and coolant pipes. One cassette covers the toroidal area of 7.5°, and 3 cassettes are replaced from a lower maintenance port (total 16 ports). The cassette design is consistent with reduction in the fast neutron flux to protect the vacuum vessel and replacement of the inner and outer heat exhaust units of the CuCrZr pipe. The cassette structure consists of F82H, and the total thickness of 25 cm can reduce the fast neutron flux efficiently by arranging two lines of puddles for the pressurized water with the path length ratio of 7:3 for F82H and water, respectively. The water flow (1m/s) in the puddles removes the total nuclear heat of 0.7 MW in one cassette (totally 32 MW for 48 cassettes).
Heat transport analyses of the W-monoblock and CuCrZr-pipe was performed in the three-dimensional (3-D) modeling by using the ABAQUS finite element method (FEM) code, considering the monoblock geometry (shaped target surface is used to protect the leading edge) and the heat flux components (radiation power, neutral flux and nuclear heat as well as plasma heat load along the field line) given by SONIC and MCNP-R simulations. Maximum temperature on the W-surface appears near the downstream edge in the plasma-wetted area. The critical operation temperature of 1200°C, i.e. W-recrystallization, corresponds to the total peak heat load of 13.5 MWm-2, which is 1.8 times higher than the result simulated by SONIC (7.5 MWm-2). The maximum temperature of the CuCrZr pipe is 351□C, and mechanical toughness of the cooling pipe is also near critical against thermal fatigue. Elasto-plastic analysis of the displacement and thermal stress on the W-monoblock and CuCrZr pipe under the higher heat load have been performed, and the results are presented.

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[4] A possible divertor combined the advantages of supper-X and snowflake for CFETR/DEMO

Presenter: DU, Hailong (Southwestern Institute of Physics)

Abstract
The experimental and modeling have shown that the advanced snowflake divertor can well mitigate heat flux loading onto target surfaces due to the smaller perpendicular incident angle and larger magnetic expansion compared with conventional divertor [1,2]. But, the large magnetic expansion of snowflake may lead to a serious problem of particle exhaust, which results in the core plasma density out of control. In order to well control particle flux, we propose a possible divetror, which combines the advantages of supper-X[3] and snowflake[4], including long leg, large magnetic expansion, closed contracture and small perpendicular incident angle. In this work, we will employ the authoritative edge plasma code SOLPS-ITER [5] to evaluate the control capacity of particle and heat flux for this divertor. The primarily modeling results from SOLPS-ITER show that the carbon impurity and recycling particles can well be screened in outer divertor region accompanying with a large quantity of radiation loss power. Moreover, the neutral pressure is very high near target region due to the good screening, so that the neutral recycling and impurity particles can be removed by the pumping system. This possible divertor not only can well control heat flux, but also remove the partial impurity and recycling particle to control particle by pumping. Such a divertor may potentially provide a power and particle handling solution for long pulse advanced tokamaks.

Reference

[1]Zheng G Y, Cai L Z, Duan X R, Xu X Q, Ryutov D D, Cai L J, Liu X, Li J X and Pan Y D 2016 Investigations on the heat flux and impurity for the HL-2M divertor Nucl. Fusion 56 126013
[2]Chen Z P, Kotschenreuther M, Mahajan S and Gerhardt S 2018 A study of X-divertor in NSTX-U with SOLPS simulations Nucl. Fusion 58 036015
[3]Rozhansky V, Molchanov P, Veselova I, Voskoboynikov S, Kirk A, Fishpool G, Boerner P, Reiter D and Coster D 2013 Modeling of the edge plasma of MAST Upgrade with a Super-X divertor including drifts and an edge transport barrier Plasma Phys. Control. Fusion 55 035005
[4]Du H and Zheng G 2019 Predictive modeling of detachment cliff with X-Divertor geometry in HL-2M by SOLPS (submitted to Nucl. Fusion)
[5]Meier E T, Goldston R J, Kaveeva E G, Makowski M A, Mordijck S, Rozhansky V A, Senichenkov I Y and Voskoboynikov S P 2017 Drifts, currents, and power scrape-off width in SOLPS-ITER modeling of DIII-D Nucl. Mater. Energy 12 973–7

[45] Status of Divertor/SOL modelling in PROCESS

Presenter: MULDREW, Stuart (CCFE, UKAEA)

Systems codes, such as PROCESS, model all systems of a power plant to investigate large numbers of design points. They are used for scoping studies and to identify areas of feasible design points.

Multi-dimensional modelling of the plasma Scrape-Off-Layer (SOL), divertor and seeded impurities is too computationally intensive to be incorporated directly into a systems code. Divertor protection parameters such as $P_{\rm sep}/R_{\rm 0}$ and $P_{\rm sep}B_{\rm T}/q_{\rm 95}AR_{\rm 0}$ have been used as a constrait for capturing the divertor problem in previous studies instead. A 1-D SOL/divertor model has been implemented in PROCESS to try and produce more accurate information regarding the divertor conditions. The aim of the 1-D model is to determine if the divertor is detached, whether the power crossing the separatrix is consistent with required conditions at the target, and to model the loss of power and momentum along the 1-D flux tube.

The following physical processes are included: convected heat flux; thermal conduction; momentum conservation; radiation by deuterium, tritium and impurities; charge exchange; electron impact ionisation; and surface recombination. Pumping is not included – all particles striking the target are recycled. The strong shearing of the flux tube near the X-point is not taken into account.

As the seeded impurity concentration is increased a discontinuous transition is observed between an attached state, where the plasma temperature at the target is $50\, \rm{eV}$, and a state where the temperature at the target hits the lower bound of the simulation, $1.1\, \rm{eV}$. We interpret this as a detached state, within the limitations of the model.

The 1-D model has been compared to 2-D models (e.g. the Japanese code SONIC) for European DEMO-like machines. However, a large database of DEMO-like runs using the detailed codes is not readily available, so benchmarking the 1-D model against detailed codes is an ongoing process.

PROCESS now also allows for a double-null divertor configuration as an alternative to the single-null considered standard for conventional aspect ratio tokamaks, but to achieve worthwhile power sharing between the upper and lower divertors would seem to require a high degree of control of the plasma position.

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 and 2019-2020 under grant agreement No 633053 and from the RCUK [grant number EP/P012450/1]. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

[50] Study of Single Null divertor in DTT with Nitrogen, Neon and Argon seeding

Presenter: RUBINO, Giulio (University of Tuscia)

The EUROfusion roadmap [1] has recognized the exhaust of large heat loads as one of the most critical issue to solve for the generation of electrical power with a Demonstration Fusion Power Plant (DEMO) by 2050. This condition is particularly challenging in the material facing the plasma in the divertor, where detached conditions must be guaranteed for safe operations of the machine. According to the limit imposed by the current technologies, the maximum tolerable peak heat flux is 10 MW/m2 and the temperature should be below 5 eV to avoid tungsten sputtering [2]. Considering the typical conditions of DEMO, this means that almost 90% of the power entering the SOL and diverted towards the divertor targets should to be radiated by inserting external impurity, as in high radiating scenarios in present experimental machines [3].
In order to bridge the gaps between present machines and the plasma conditions in DEMO, the Divertor Tokamak Test Facility (DTT) is in construction in Italy. The main goal of DTT is to test alternative divertor solutions - as Advanced Divertor Configurations (ADC) and liquid metal divertor – in DEMO relevant regimes [4]. The possibility to adopt these solutions will be assessed in terms of compatibility with core performances and of technological feasibility considering both materials and engineering aspects.
In this work we present the power exhaust analysis of the divertor for the conventional Single Null magnetic configuration in DTT. The study is performed by means of the 2D edge codes SOLPS-ITER [5] and SOLEDGE2D-EIRENE [6]. We refer to the medium density scenario characterized by a separatrix density of ne,sep = 5 x 1019 m-3 and an external input power level of Padd = 45 MW since it is the baseline scenario used to compare all the other alternative divertor solutions. The injection of nitrogen, neon and argon have been considered. In order to define the effect of the different impurities on the plasma performances and to study DTT operational scenarios, a scan in the impurity puffing rate has been performed. Here, we present a discussion on the possible operational windows in terms of tolerable heat loads, achievement of the detachment regime and radiating power fraction as a function of the different N, Ne and Ar puffing levels. Finally, the influence on the plasma performances on the choice of the impurity species is assessed by considering impurity concentrations in the three cases.

[1] European Research Roadmap to the Realisation of Fusion Energy, November 2018, https://www.euro-fusion.org/fileadmin/user_upload/EUROfusion/Documents/2018_Research_roadmap_long_version_01.pdf
[2] G. van Rooij et al., J. Nucl. Mater. 438 (2013) S42.
[3] Bernert M., (2017) Nuclear Materials and Energy 12 111–118 ISSN 2352-1791
[4] DTT Divertor Tokamak Test facility Interim Design Report, April 2019
[5] S.Wiesen, et al., (2015) Journal of Nuclear Materials, 463, 480-484
[6] Bufferand, H., et al., (2015) Nuclear Fusion, 55, 053025

[11] Assessment of the pumping efficiency in DEMO conventional and alternative divertor configurations

Presenter: IGITKHANOV, YURI (Karlsruhe Institute of Technology)

The design of a DEMO divertor is an important task which defines the reflux of fuel and helium neutral particles to the plasma and finally determines the particle exhaust and pumping efficiency. For the conventional divertor, optimization of the dome height and its effect on neutral compression, position of the pumping ports as well as the effect of neutral gas screening by electrons in the private flux region (PFR) are analyzed. The analysis includes the calculation of neutral density in the sub-divertor and the overall conductance of the sub-divertor structure, which consequently affects the estimation of the effective pumping speed and the achievement of detachment. The divertor configuration with dome impedes the reflux of neutrals towards the plasma through the x-point. The screening of the neutrals from backflow to the Scrape-off layer (SOL) by plasma electrons at the separatrix is estimated.
Alternative configurations for the DEMO divertor are being explored to achieve an improved mitigation of the heat and particle loading at the plasma-material interfaces. These configurations with a variable volume in the PFR, which consequently influences the neutral behavior and hence the achieved divertor pumping efficiency. This paper studies, in terms of the pumping efficiency, a wide range of prominent proposed alternatives to the conventional, single-null divertor, namely the “double null” divertor, the “X divertor“, the “Super-X divertor” and the “snowflake” divertor.
The investigation of the impact of neutral gas dynamics on the particle exhaust for different divertor configurations under steady-state operation is performed using the numerical tools DIVGAS (Divertor Gas Simulator) and ITERVAC. The DIVGAS code is based on the Direct Simulation Monte Carlo (DSMC) method, in which the solution of the Boltzmann kinetic equation is reproduced by simulating the collisions and the ballistic flight of model particles, which statistically mimic the behavior of real molecules. The ITERVAC code is a semi-empirical code which models divertor configurations as a network of connected channels and is appropriate for quick engineering calculations. In our workflow, the neutral flow field for each of the divertor configurations is derived from plasma boundary conditions along the separatrix, extracted from corresponding plasma simulations, exactly at the interfaces between the SOL and the PFR. All assumed plasma scenarios are based on a highly dissipative divertor relying on a partially detached divertor operating regime, similar to ITER, or even on full detachment. Moreover, the position and the size of the pumping port as well as the influence on the neutral flow behavior in the PFR are analyzed and the advantages and disadvantages of each case are discussed.
This work is complementary to the overview papers on the conventional divertor and on the alternative configurations.

[53] Edge and divertor modelling of JT-60SA ITER-like scenario with carbon wall

Presenter: LUCA, Balbinot (Consorzio RFX)

The main objective for JT-60SA is to study magnetically confined plasma in near-fusion scenarios in support of ITER and DEMO. One of the major open issues is to demonstrate divertor heat and particles handling in ITER-like plasma conditions. One of the options for JT-60SA divertor in the Integrated Research phase II is to use Tungsten as plasma facing component[1], while in the initial research phase it will be equipped with full carbon walls. JT-60SA ITER-like scenario is an inductive single null divertor configuration with 41MW of input power, the maximum available for the machine, and presents the major challenge in terms of power handling because of its low density at the separatrix $n_e=1\times 10^{19}m^{-3}$ [1]. Similar studies have been made for lower input power JT-60SA scenarios [2], while the 41MW scenario has been simulated with higher densities with the integrated divertor simulation code SONIC[3].
JT-60SA full carbon divertor mono-block allows heat loads up to 15MW/m^2. Edge modelling by means of coupled plasma-neutral codes is needed to predict what operational regimes can be handled by the divertor: such codes indeed are the most reliable to calculate particle and heat transport in high recycling and detachment conditions. SOLEDGE2D-Eirene is particularly suitable for this case for its ability in describing plasma-wall interaction for the proposed magnetic configurations. By means of SOLEDGE2D-EIRENE, in this work, we have evaluated if, for the JT-60SA high power low-density scenario, the radiation fraction foreseen in the SOL (higher than 85% [1]) is achieved by the intrinsic carbon impurities and the corresponding core contamination. We have also studied the effect of additional seeded impurities, and whether they can provide an advantage in respect of the pure deuterium solution.
In order to define reasonable transport parameters for modelling, we have considered JET pulses with carbon wall and with global parameters (such as heating power, n/n$_G$, n$_{core}$/n$_{sep}$, plasma shape and strike point positions) as much close as possible to those of the analysed scenario. Edge transport parameters have been evaluated in such compatible condition by comparing SOLEDGE2D-EIRENE results to experimental data from spectrometers, Langmuir probes and Thompson scattering. The model has then been applied to JT-60SA scenario. With such a model, the edge plasma parameters of scenario have been estimated, the heat load to the divertor and the impurity influx needed to reduce the heat flux to the divertor under steady-state conditions have also been calculated.

[1]JT-60SA Research Plan: http://www.jt60sa.org/pdfs/JT-60SA_Res_Plan.pdf
[2]M.Romanelli et al., Investigation of Sustainable Reduced-Power non-inductive Scenarios on JT-60SA, (2017)
[3]H.Kawashima et al., Evaluation of heat and particle controllability on the JT-60SA divertor, J. Nucl. Mater. 415 (2011)

[64] Optimization of the impurity seeding recipe in terms of power dissipation, core radiation and fuel dilution with Ar and N seeded SOLPS 5.0 simulations for ASDEX Upgrade

Presenter: HITZLER, Ferdinand (Max-Planck-Institut für Plasmaphysik)

In a next step fusion device like ITER or DEMO, the unmitigated power loads at the divertor targets will considerably exceed the allowed material limits which are foreseen to be in the order of $5-10\,\mathrm{MWm}^{-2}$. Therefore, to prevent severe damage of plasma facing components and erosion of target material, a significant amount of power has to be exhausted via impurity radiation. For this purpose, it will be important to identify an optimum seeding recipe which provides sufficient power dissipation, but at the same time only has a minimal impact on the confined plasma and the burn conditions. Therefore, in this contribution argon and nitrogen seeding is investigated and compared via SOLPS 5.0 modeling of ASDEX Upgrade H-mode plasmas. Impurity seeding scans are performed in which nitrogen shows considerably less core radiation compared to the argon seeding case at comparable divertor conditions (i.e., at similar target temperatures and peak power loads). However, at the same time nitrogen seeding leads to stronger fuel dilution. These properties can be explained by the different impurity radiation efficiencies and differences in the impurity density distributions. A trade-off between core radiation and fuel dilution can be achieved by mixing both impurities. A previous analysis of the impurity transport and the divertor retention revealed that the reaction of the main ion plasma flow on the impurities differs for argon and nitrogen which will be discussed in more detail. In addition to the impurity seeding scans, a scan of the input power is performed. Preliminary studies show a very different reaction of the plasma on the impurity seeding at higher input power which motivates a closer analysis of these high-power simulations.

[55] Electromagnetic and mechanical analysis of alternative magnetic divertor configurations for DEMO

Presenter: AMBROSINO, Roberto (Consorzio CREATE)

The development of a reliable solution for the power and particle exhaust in a reactor is recognized as a major challenge towards the realization of DEMO [1]. Alternative magnetic configurations such as Double Null, Snowflake, X and Super-X divertors are considered as a promising solution to reduce the heat load on the divertor targets even if their scalability on a DEMO size device is a challenging engineering problem.

The definition of an alternative magnetic configuration requires an optimization of the machine geometry, from the plasma facing components to the divertor structures; dedicated solution for the toroidal and poloidal coil systems are also needed to reduce the vertical forces and mechanical stresses on the active structures. Additional engineering limitations are related to remote maintenance and the plasma controllability of the alternative divertor concepts.

In this paper, starting from the results in [2], an electromagnetic and mechanical analysis of DEMO alternative configurations in presented. The controllability of the plasma configuration is tackled in terms of plasma vertical stability and shape sensitivity respect to a prescribed set of disturbances. Finally, the possibility to increase the range of alternative divertor concept considering also double null alternative divertor concepts is investigated assuming the presence of in-vessel divertor coils with a current limitation to 1MAturns.

[1] Fusion Electricity – A roadmap to the realisation of fusion energy, November 2012 (http://users.euro-fusion.org/iterphysicswiki/images/9/9b/EFDA_Fusion_R oadmap_2M8JBG_v1_0.pdf)
[2] R. Ambrosino, A. Castaldo, S. Ha, V.P. Loschiavo, S. Merriman, H. Reimerdes, Evaluation of feasibility and costs of alternative magnetic divertor configurations for DEMO, Fusion Engineering and Design, 2019.

[35] First multi-fluid modelling results of super-X divertor in DEMO with Ar seeding

Presenter: XIANG, Lingyan (CCFE)

As a large fusion machine with high power density, DEMO requires sufficient capability of power handling at its boundary region. Apart from applying impurity seeding to actively dissipate power via radiation to cope with this demand, exploring divertor configurations that may be more advantageous than the conventional single-null in handling high power flux crossing the separatrix is also desired to lay the foundation for DEMO divertor design.

The Super-X divertor (SXD) configuration features large plasma wetted area due to large total flux expansion as well as large Rdiv, which is very preferable to achieve lower peak heat flux at the target. It also has long midplane to target connection length, which further increases the wetted area due to longer time for cross field transport of particles and heat. This configuration, owing to its well baffled long divertor leg, should give good divertor closure for neutrals. Namely neutrals do not easily leak from the divertor leg to the main plasma. SXD also has large divertor volume which means potentially higher maximum radiated power from the divertor. Besides, due to the aforementioned features, the plasma in the divertor is cooler, meaning higher efficiency for volumetric losses. These features make SXD an interesting configuration to be explored in DEMO.

This contribution will present the first results of an exploration into argon seeded DEMO plasma in SXD configuration, using the multi-fluid plasma boundary modelling code SOLPS-ITER. In the study, engineering parameters like the deuterium fuelling and argon seeding rates are scanned, inducing a broad range of divertor plasma conditions. Hence, it allows for a systematic investigation of divertor detachment, i.e. the onset, the parameter window, the stability etc., in the SXD configuration in DEMO environment. The parameter scans also enable insights into how the radiation distribution and the enrichment of argon impurity in the DEMO SXD dictates/is dictated by the divertor plasma conditions. On the other hand, with increased connection length from midplane to the outer target, the partition of Psep at the low-field-side midplane between the inner and outer divertor will change. At some point, the connection length to the outer target may get long enough to make the inner divertor a concern of power exhaust, now receiving more power. This is also looked at in this contribution.

[2] R-matrix calculations of electron-impact excitations of Ne-like W LXV

Presenter: BARI, Muhammad Abbas (Pakistan Atomic Energy Commission)

Abstract. The current design of the International Thermonuclear Experimental Reactor (ITER) diverter calls for tungsten to be employed for certain plasma facing material in the diverter region. Electron impact excitation (EIE) and radiative rates of tungsten ions are basic atomic processes in nuclear fusion plasmas of the ITER tokamak. Therefore, accurate electron collisional data are required for emission modelling of tungsten ions. Electron-impact excitation for Ne-like W64+ is of particular importance, since tungsten L-shell ions (Ne-like W64+– Li-like W71+) will be abundant in ITER core plasmas. In this paper, we present new large scale R-matrix (up to n=5) scattering calculations for the electron collisional excitations of Ne-like W64+. The Maxwellian averaged effective collision strengths for these transitions are computed over a wide temperature range (log10Te(K) = 3.0–8.0) for various plasma conditions. The rates from these parallel intermediated-coupling frame transformation (ICFT) R-matrix method atomic calculations will be employed for collisional-radiative modelling of this ion. The resonance contributions to effective collision strengths are dominant at lower temperatures and can significantly affect the line intensities in our spectral modelling.

[60] Impact of impurity seedings for divertor protection against intolerable heat loads and tungsten sputtering on general on plasma performances using the SYCOMORE system code

Presenter: KAHN, Sebastien (UKAEA)

The next step after ITER is the demonstration of stable electricity production with a fusion reactor. Key design performances will have to be met by the corresponding power plant demonstrator (DEMO), fulfilling a large number of constraints. System codes such as SYCOMORE, by simulating all the fusion power plant sub-systems, address those questions. SYCOMORE uses an advanced two points model to simulate the scrape-off layer (SOL) physics, that takes momentum losses and impurity radiation into account. As impurity radiation affects both the core and the SOL power balance, a loop between the SOL and the Core models is designed to find the minimal impurity fractions necessary to protect the divertor targets from both intolerable heat flux per unit of surface (qpeak) and tungsten sputtering (maximum target plasma temperature). This coupling allows to address the effect of divertor protection on gobal power plant design. Sensitivity analysis can be then used to compare the effect of the divertor design and physics uncertainties on global power plant design with uncertainty sources from other part of the tokamak. Such analysis will be performed starting from the ITER design, comparing the effect of the uncertainty on the upstream SOL width parameter (lambaq), to the following other uncertainty sources: toroidal magnetic field on axis (BT), energy confinement enhancement factor (fH), electronic density at the separatrix (nsep) and auxiliary plasma heating power (Padd). The effect of the divertor target plasma temperature constraint (Ttarg, max) will be compared to the divertor target heat flux per unit of surface (qpeak, max) one by performing the same sensitivity analysis using alternatively the qpeak, max constraint and the Ttarg, max one. The results of these two analyses will be compared to a more realistic one that take the two divertor target constraints altogether.

[70] The Divertor Tokamak Test facility

Presenter: MARTIN, Piero (University of Padova and Consorzio RFX, Padova, Italy)

Appropriate disposal of the non-neutronic energy and particle exhaust in a reactor is universally recognized as one of the high priority challenges for the exploitation of fusion as an energy source. The new Divertor Tokamak Test (DTT) facility, which will be built in Italy, is a tool to address that challenge in high-field, high performance tokamak with complete integration between core and edge plasma scenarios.

This paper will present a review of the present status of DTT design.

The Divertor Tokamak Test facility is a superconducting tokamak with 6 T on-axis maximum toroidal magnetic field carrying plasma current up to 5.5 MA in pulses with total length (including start-up, flat-top and ramp-down phases) up to 100 s. The D-shaped device is up-down symmetric, with major radius R=2.14 m, minor radius a=0.64 m and average triangularity 0.3. The auxiliary heating power coupled to the plasma at maximum performance is 45 MW, shared between the three heating systems used in ITER and foreseen in the present version of DEMO: ion and electron cyclotron resonance heating and negative ion beams. In particular DTT will use 170 GHz ECRH, 60-90 MHz ICRH and 400 MV negative ion beam injectors. DTT will start operation with 8 MW ECRH, which will increase to 25 MW within two years from the beginning of plasma operation. The full 45 MW heating power is planned to be reached within 6 years from the beginning of operation. An input power of 45 MW allows matching the PSEP/R values, where PSEP is the power flowing through the last closed magnetic surface, with those of ITER and DEMO. DTT is in fact designed to reach PSEP/R =15 MW/m. Fast particles up to 700 keV (simulating the alfa generated by nuclear fusion reaction) will be generated through ICHR power while the 400 keV NNBI allows for producing a super-Alfvenic population to study fast particle collective transport and wave particle interaction.

DTT is being designed with a high level of flexibility, in particular as far as divertor scenarios are concerned. From a magnetic point of view the external coils together with a set of four internal coils will allow to control and optimize the local magnetic configuration in the vicinity of the divertor target. The main divertor magnetic topologies, which can be produced in DTT are the reference single null, double null and snowflake configurations. These can be produced at (or close) to the maximum target plasma current of 5.5 MA, while double super-X may be feasible only at significantly lower current (and with vertical stability issues still to be solved).

The DTT poloidal field coils system also allows for the realization of scenarios with negative triangularity. In particular a 5 MA single null scenario with delta=-0.13 and delta_lower=-0.16 and a double null scenario at 3.5 MA with delta=-0.38 can be produced.

[26] Simulation study of the radiative divertor of different seeded impurity species for CFETR

Presenter: MAO, Shifeng (University of Science and Technology of China)

The fusion power of China Fusion Engineering Test Reactor (CFETR) [1] is proposed to achieve the level of gigawatt, which implies the critical issue of power exhaust by divertor. Impurity radiation is an effective way to reduce the heat load onto the divertor target. For CFETR, to reduce the tritium retention and increase the lifetime of plasma-facing components, full-tungsten wall would be the prior choice, which means there is not any intrinsic radiative impurities. Therefore, impurity seeding is indispensable for CFETR. It is necessary to explore the reasonable impurity seeding scheme to achieve good performance of radiative divertor, where a high radiation fraction can be achieved to reduce the heat load and the upstream impurity concentration can be kept as low as possible to avoid avoid degradation of the performance.
In this work, SOLPS simulation are performed for the radiative divertor with different radiative impurity species, including nitrogen, neon and argon. For a radiation fraction of ~85%, the boundary plasma is simulated with different density. A comparison of the radiative efficiency H, which is defined by the radiative fraction divided by Zeff-1, shows HN>HAr>HNe. Further analysis shows that, the residence time is longer for the impurity of higher ionization potential, which implies the non-coronal effect is more significant for the impurity species of lower ionization potential. The ne, Zeff and Prad from SOLPS simulations are then fitted according to a modified Matthews' form [2], where the influence of ionization potential of impurity species is included.

Reference
[1] Y.X. Wan, et al., Nucl. Fusion 57 (2017) 102009.
[2] G.F. Matthews, et al., J. Nucl. Mater. 241-243 (1997) 450-455.

[21] Advanced Power Exhaust Studies for New Lower Tungsten Divertor of EAST under High Power and Steady State Operations

Presenter: SI, Hang (Institute of Plasma Physics Chinese Academy of Sciences)

Divertor is one of the key components in Tokamak. The control of heat flux and erosion of the divertor target is one of the grand challenges facing the design and operation of next-step high-power steady-state fusion. It is essential to efficiently dissipate power in the divertor to ensure the maximum steady-state power load at the divertor target below 10 ~ 15 MW/m^2. In addition, adequate reactor lifetime dictates near zero-erosion at solid PFCs, so the electron temperature at the divertor target plates must be maintained at a low temperature with Te ≤ 5 ~ 10 eV to suppress erosion.

In response to this challenge, recently Experimental Advanced Superconducting Tokamak(EAST) has launched a new initiative to develop and validate the advanced divertor concepts for the design of new lower tungsten divertor. In general, developing an advanced divertor configuration requires: (1) Increasing divertor closure by divertor baffling to improve divertor screening for recycling neutrals and impurities, hence increasing divertor neutral pressure, thus enhancing divertor particle and power exhaust. Several divertor configurations with various target inclination angles are modelled to evaluate the effect of divertor closure on detachment. The modeling results show that increasing divertor closure can significantly trap more neutrals with the same upstream separatrix density and hence decrease the onset of detachment. Moreover, with increasing divertor closure the divertor radiated power is also increased and the peak heat flux density at the divertor target is reduced. (2) Optimizing magnetic configuration to extend the plasma wetted area through flux expansion, and increasing the divertor volume by increasing the field line length. Based on the flexible poloidal field control system in EAST, some alternative advanced magnetic configurations, i.e., quasi snowflake (QSF)[1] , aka X-divertor and fishtail divertor (FTD)[2] have been attempted for the new lower tungsten divertor. According to the modeling results, QSF and FTD can significantly reduce the peak heat flux density at the lower outer target by the magnetic flux expansion and outer strike point sweeping respectively. Furthermore, an alternative advanced divertor coupling divertor closure with advanced magnetic configuration (QSF or FTD) in EAST can further facilitate new lower tungsten divertor detachment, which may provide a promising means for the design of advanced divertors in future fusion devices.

References:
[1]G. Calabro, et al, 2015 Nucl. Fusion 55 083005
[2]X.D. Zhang, et al, 2nd IAEA Technical Meeting on Divertor Concepts, Suzhou, China, 13-16 November 2017

[69] Scoping study of dissipative divertor scenarios for SPARC

Presenter: UMANSKY, Maxim (Lawrence Livermore National Lab)

Operating at 12 tesla on axis with a plasma current of 7.5 MA and total fusion power of 100 MW, SPARC [1] is projected to have a power exhaust heat flux width of 0.2 mm with an unmitigated parallel heat flux of up to 30 GW m-2 entering the divertor. While recent UEDGE modelling of other high-field tokamaks designs, ADX and ARC, indicates that long-leg divertors can dramatically improve divertor power handling – producing fully detached solutions with benign power fluxes to material surfaces [2,3] – SPARC does not have a neutron shield blanket and therefore only has limited space for the divertor legs. Consequently, the current approach for mitigating target heat fluxes is to take advantage of whatever divertor dissipation may be available and rapidly sweep the strike point over a large divertor target area while employing large target plate tilt angles.In this present study, we use UEDGE to examine the level of divertor dissipation that may be obtained in SPARC for representative divertor leg lengths and target plate geometries. UEDGE is set up to match the SPARC design dimensions and projected upstream plasma parameters and the exhaust power flux; then the numerical solutions for the divertor are analyzed varying details of the divertor geometry (the leg length, target plate tilting) and plasma physics assumptions (anomalous transport in the leg, impurity ion radiation) to determine whether fully- or partially-detached divertor plasma scenarios exist and at what power exhaust levels.

[1] Greenwald et al. “The High-Field Path to Practical Fusion Energy” MIT PSFC report PSFC/RR-18-2 (2018); [2] Umansky et al., Phys. Plasmas 24 (2017) 056112; [3] Wigram et al., Contrib. Plasma Phys., 58 (6–8) (2018), p. 791

Programme Committee Meeting - Board Room C (C Building, 4th Floor) (6 Nov 2019, 16:30-17:00)

Dinner (6 Nov 2019, 19:00-21:00)

Dinner at Gasthaus Möslinger (Stuwerstraße 14, 1020 Wien)

Thursday 07 November 2019

Divertors in Next Step Devices - Board Room C (C Building, 4th Floor) (7 Nov 2019, 09:00-10:10)

time [id] title

09:00 **[82] The first ITER tungsten divertor: what do we hope to learn?**

Presenter: PITTS, Richard (ITER Organization)

On the eve of component procurement and with a substantially updated Research Plan [1] describing the pathway to achievement of inductive and steady state burning plasma operation on ITER, the present physics basis for the first ITER tungsten (W) divertor is outlined, focusing on the main design and operational driver: steady state and transient target power fluxes [2]. With the dimensions and shaping of the W monoblocks (MB) constituting the vertical targets (VT) now fixed, the operating space in allowable power flux density is more readily constrained, both from the physics and materials sides. The combination of MB front surface shaping and VT tilting for misalignment protection, fluid drifts and potentially very narrow near SOL heat flux channels (λq), push the operating space up to higher sub-divertor neutral pressures. This takes the divertor closer to full detachment, possibly approaching limits on both upstream density and pedestal radiation and presenting detachment control issues. A revised criterion for maximum tolerable loads based on the avoidance of W recrystallization [3] is being substantiated by expanded materials studies and does provide some room for manoeuvre on ITER, at least for the first divertor, for which the required lifetime is expected to be around 2000 hours of burning plasma. Together with the existing database of SOLPS-4.3 simulations [2], the newest simulation results obtained with the SOLPS-ITER code package will be presented, including the most recent database of drift simulations and studies with reduced heat transport to examine the cost of low λq with extrinsic seeding. Work is now being performed to examine the potential gain of poloidally expanded equilibria, using new scenarios with the maximum such expansion possible at high current. A detailed divertor simulation study is also underway focusing on the much lower performance discharges which will characterize the early years of ITER operation.

Assessments have been made of the pellet and fuelling requirements compatible with control of detachment, core plasma W accumulation and H-mode operation at different phases of the Research Plan. Limits on allowable ELM energy losses are fixed both by accumulation and transient radiation under the assumption of no ELM-induced MB surface melting, but the real limit is likely to be fixed by the W material itself (surface cracking), or by the inevitable castellation edges which appear in any target comprising discrete heat flux handling elements. All of which drives the need for ELM suppression, the principal method for which on ITER is planned to be through 3D magnetic perturbations. A simulation programme is in place to examine the possibility for divertor heat load dissipation at high performance on ITER in regimes in which ELM suppression is expected.

In describing the current status of the first ITER divertor physics basis, the presentation will attempt to discuss what can be learned with respect to devices beyond ITER. It will also highlight the many areas in which more R&D is required to improve understanding for ITER in the remaining years before divertor operation begins.

[1] "ITER Research Plan within the Staged Approach", ITER Technical Report ITR-18-003 available at: https://www.iter.org/technical-reports

[2] R. A. Pitts et al., Nuclear Materials and Energy, in press.

[3] G. De Temmerman et al., Plasma Phys. Control. Fusion 60 (2018) 044018.

[4] A. R. Polevoi et al., Nucl. Fusion 58 (2018) 056020.

09:30 **[52] Recent progress on divertor physics design of CFETR**

Presenter: DING, Rui (Institute of Plasma Physics, Chinese Academy of Sciences)

Since 2017, significant efforts have been made by Chinese community for the physics and engineering design of Chinese Fusion Engineering Testing Reactor (CFETR) [1], which is proposed to bridge the gap between ITER and DEMO. One of the key challenges is that the divertor solution for CFETR must meet requirements beyond that of ITER. For the standard CFETR operation with the fusion power up to 1 GW, the power across separatrix per unit major radius can reach 30 or 25 MW/m with respect to either fully non-inductive or hybrid mode operation, which are much higher than 17 MW/m for ITER. The high duty cycle of CFETR (0.3-0.5) requires negligible divertor target erosion rate and low enough heat loads that plasma-facing components are capable of withstanding. The divertor solution also needs to be compatible with high core fusion performance, which means low impurity contamination and efficient helium exhaust.

The design of divertor geometry has a strong correlation with the plasma configuration and also needs to meet the requirements of the first-wall and blanket. For the plasma current of 13.78 MA, the snowflake configuration could not be obtained since the current of divertor superconducting coil exceeds the engineering limits. Therefore, a lower single-null equilibrium based on the steady state scenario has been used for the divertor design. The configuration has a lower triangularity of 0.46 for a better divertor volume optimization. A conventional divertor configuration with long divertor legs (1.7 m for outer leg and 1.3 m for inner leg) and V-shaped design at the two target corners have been proposed for the referenced configuration. Since tungsten (W) will be used as the divertor armour material, extrinsic impurities need to be introduced as the main radiators to reduce the divertor heat loads. The stationary heat loads on divertor targets have been analyzed from simulations using the SOLPS code with seeding impurities neon (Ne) and Argon (Ar). With the same injection rate of 5×1020/s, the peak heat loads are reduced much lower than the limit and the total radiation power is similar for both impurities. However, Ne radiation mainly occurs within the divertor volume, and thus shows better compatibility with the core plasma. Although detachment is not obtained at the far SOL region, the low particle flux still leads to acceptable W erosion rate there.

In addition, an even longer divertor leg configuration (2.4 m for outer leg and 1.6 m for inner leg) and a small angle slot configuration (SAS) with normal leg lengths [2] have been studied in comparison to the referenced configurations. Both of the two configurations show better radiative abilities in the divertor for the same upstream conditions, and therefore, leads to lower heat flux and plasma temperature on the targets. Furthermore, BOUT++ has been used to simulate the ELMs behaviours, which helps to evaluate the transient heat loads on the divertor targets.

[1] G. Zhuang, et al., Nuclear Fusion 59 (2019) 112010
[2] H.Y. Guo et al., Nuclear Fusion 57 (2017) 044001

09:50 **[54] A strategy to develop power exhaust solutions for tokamaks beyond ITER**

Presenter: CANIK, John (Oak Ridge National Laboratory)

Predictions for the scrape-off layer (SOL) in future fusion devices based on empirical scalings imply extremely large parallel heat flux, q|| ~10 GW/m2, which is exacerbated for high-field concepts that may enable a Compact Pilot Plant (CPP) as recommended by a recent US strategic planning assessment. Here we discuss the framework of a proposed ~10 year program to more firmly establish the basis for power handling in tokamaks by extending the paradigm of solid (likely high-Z) walls using noble gas seeding to increase impurity radiation. This is intended to take relatively mature science and technologies relevant to power exhaust and particle control (PFCs, pumping) and demonstrate the combined physics and engineering basis for a divertor solution for a tokamak-based CPP. The aims of this program are to a) establish the predictive basis for projecting heat flux and the conditions for its mitigation; b) develop core confinement scenarios that minimize exhaust requirements on the divertor; c) test the interaction between radiative divertor and high-pressure pedestals at-scale; and d) explore the potential of alternative divertor geometries as risk mitigation should the conventional geometry employed in ITER prove to be insufficient for CPP requirements.
Although there are important distinctions, many of the goals of a CPP are similar to other tokamak-based pathways (e.g. EU-DEMO, CFETR), which should allow for collaborative pursuit of power exhaust solutions. Significant progress has been made in understanding the SOL heat flux width, and recent studies have focused on how the solution to the heat flux challenge scales, for example through the impurity fraction fZ required for detachment [1,2]. Developing and confirming these predictions is essential, especially for CPP-scenarios which may stress higher-field approaches. It is also necessary to develop exhaust scenarios that are compatible with core confinement requirements. With the operation of burning plasma experiments on the horizon via ITER and SPARC, the compatibility of the divertor and pedestal can be studied nearly at-scale, with reactor-like dimensionless and dimensional parameters. In parallel, new core scenarios should be developed that demonstrate tradeoffs between sustaining high confinement in the presence of strong core radiation versus raising PSOL. While multiple members of the worldwide program are actively developing alternative divertor geometries these focus primarily on the high poloidal flux expansion branch of advanced divertors, e.g. snowflake and x-divertors. A new experimental capability is emphasized as being necessary to explore the physics of long-legged, tightly baffled divertors at high heat flux, testing for example the predicted access to new regimes with dominant roles of divertor turbulence [3] that may dramatically lessen the power exhaust challenge.

[1] M.L. Reinke, Nucl. Fusion 57 (2017) 034004.
[2] R.J. Goldston, M.L. Reinke and J.A. Schwartz. PPCF 59 055015 (2017)
[3] M.V. Umansky et al, Phys. Plasmas. 24 (2017) 056112.

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Coffee Break - Board Room C (C Building - 4th Floor) (10:10-10:30)

Divertors in Next Step Devices: Continued - Board Room C (C Building, 4th Floor) (7 Nov 2019, 10:30-11:10)

time [id] title

10:30 **[83] Power exhaust studies in the Divertor Tokamak Test facility**

Presenter: VIANELLO, Nicola (Consorzio RFX, Associazione Euratom-ENEA sulla Fusione)

Dealing with power exhaust is one of the most difficulty tasks in foreseen fusion reactor based on magnetic confinement. It is commonly recognized that the standard Single Null Divertor (SND) configuration can face some difficulties in providing a scalable solution based on H-mode tokamak operation compatible with present solid divertor target technological solutions. To provide a safer solution two approaches are being considered: the adoption of alternative divertor magnetic configurations (ADCs) could favour plasma detached operations, thus mitigating the heat load on targets; liquid metallic targets are also studied for their ability to survive to high heat load while providing also a self-mitigation of incoming heat flux.

The Divertor Test Tokamak (DTT) facility [1] is being designed to specifically explore and qualify most of the ADCs (and liquid metal targets) with divertor conditions as close as possible to those foreseen in DEMO fusion reactor in terms of power crossing the separatrix, Psep/R, and heat flux decay length, □q. All of the present more promising alternative divertor configurations are realizable in DTT: the flux flaring towards the target (X divertor), the increasing of the outer target radius (Super-X) and the movement of a secondary x-point inside the vessel (X-point target) as well as the entire range of Snowflake (SFD+/SFD-) configurations [2] and the presently reconsidered double null (DND) one. This divertor configurations flexibility is supplemented by a similar flexibility in operation scenarios which for example allows scanning triangularity from the DEMO positive value to negative one or moving from H-mode to the I-mode to test possible ELMs free high confinement regimes.

To analyse the possible beneficial effects ADCs configuration in DTT and starting to optimize plasma scenario and divertor geometry, 2D edge fluid-kinetic simulations has been done on previously described divertor configurations. The analysis has shown in pure deuterium the extension of the detached operation towards higher Psol power with all the ADCs, with respect to the SND, with the best results achieved at all targets for the SFD-. Modelling has also shown that the ADCs advantages extend to the case of (Ne/Ar) impurity seeding, whereby detached operations can be reached with a lower Zeff or dilution at the separatrix compared to the SND.

References

[1] R.Albanese and H. Reimerdes, Fusion Engineering and Design 122 (2017) 285–287 https://doi.org/10.1016/j.fusengdes.2016.12.025

[2] R. Ambrosino, et al., Fusion Engineering and Design 122 (2017) 322–332, https://doi.org/10.1016/j.fusengdes.2017.01.055

10:50 **[14] The physical design of EAST lower tungsten divertor by SOLPS modeling**

Presenter: SANG, Chaofeng (Dalian University of Technology)

The divertor target is the most intense plasma-surface interaction area in the tokamak. To keep the lifetime of the device and maintain long pulse discharges, the power load and particle removal control become to be the critical issues. The lower graphite divertor of EAST tokamak is the main limitation to the achievement of further high-power long-pulse discharges [1]. To solve this problem, EAST will upgrade its lower divertor to use tungsten material. In this work, the divertor physical design is presented. The requirements for the design includes: (1) heat flux to the target below 10 MW/m2 to protect the target; (2) Te< 10 eV including the far SOL at the target; (3) W impurity control and particle removal. To this end, the new divertor should have strong power dissipation, impurity screening and particle removal capabilities. By using the 2D edge plasma code SOLPS modeling [2-4], the systematic examination of different target shapes, target angles, the pump locations, have been carried out. The optimized divertor geometry for the EAST discharge configuration is proposed. The particle exhaust, which is of essential importance for long-pulse discharges, is analyzed by the effective particle removal with different particle recycling rate at the pump. To sustain the high power discharges, Ar and Ne seeding to improve the power exhaust, and the corresponding W sputtering and W impurity transport, are studied by the coupling of the DIVIMP [5] and SOLPS. Moreover, the quasi-snowflake is assessed for better understanding of the advanced magnetic configuration application on the heat flux control. These studies can improve the understanding of power and particle exhaust, W sputtering and transport during long pulse high power operations for CFETR.

[1] X. Z. Gong et al., Plasma Sci. Technol. 19 (2017) 032001.
[2] R. Schneider et al., Contrib. Plasma Phys. 46 (2006) 3.
[3] C. F. Sang et al., Nucl. Fusion 57 (2017) 056043.
[4] C. F. Sang et al., Plasma Phys. Control. Fusion 59 (2017) 025009
[5] P. C. Stangeby and J. D. Elder, J. Nucl. Mater. 196-198 (1992) 258

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Discussion Session: Next Steps - Board Room C (C Building, 4th Floor) (7 Nov 2019, 11:10-12:00)

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

Discussion Session: Wrap up - Board Room C (C Building, 4th Floor) (7 Nov 2019, 12:00-13:00)

Discussion sessions aim at identifying the most critical issues, based on both their uncertainty and impact, and what would be the most productive path to address those issues. These findings will be compiled in a report highlighting issues and approaches to resolution for future divertor design.

Farewell and Closing - Board Room C (C Building, 4th Floor) (7 Nov 2019, 13:00-13:15)

Closing addresses by the Programme Committee and the IAEA.

-Conveners: Barbarino, Matteo (International Atomic Energy Agency); Leonard, Anthony W. (USA)

Executive Discussion - Board Room C (C Building, 4th Floor) (7 Nov 2019, 13:15-14:00)