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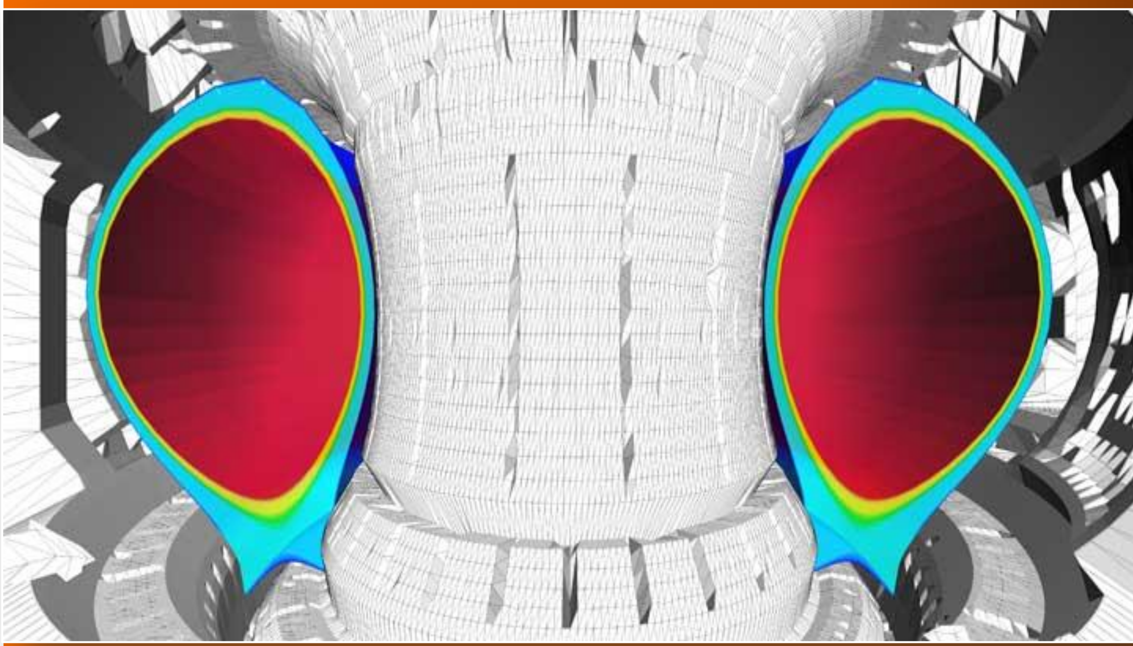
2nd IAEA Technical Meeting
Divertor Concepts

Programme

Book of Abstracts



Suzhou, China
13 – 16 November 2017

2^{nd} IAEA Technical Meeting on Divertor Concepts

13-16 November, 2017

Suzhou, China

IAEA Scientific Secretary

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Meeting Website:

https://nucleus.iaea.org/sites/fusionportal/Pages/Divertor%20Concepts/General-Info.aspx

Sessions

I. Alternative Divertor Configurations

II. Liquid Metal PFC

III. Divertor Physics and Modelling

IV. PFC Development

V. Effect of 3D Fields

VI. Confinement Vs Divertor

Schedule

Monday, 13 November

8:30-09:00	**Welcome and Opening Address** *Neu R., Gonzalez de Vicente S.M.*
Session 1: Alternative Divertor Configurations Chair: Theiler C.G.	
09:00-9:45	**It-1:** Theiler C.G. *Overview of SOL Transport and Detachment in Alternative Divertor Geometries on TCV*
9:45-10:15	**Iv-1:** Umansky M.V. *Simulations of Passively Stable, Fully-Detached Regimes in Long-Legged Divertor Configurations*
10:15-10:45	**Iv-2:** Guo H.Y. *Taming Plasma-Materials Interface for Steady-State Fusion*
10:45-11:15	**Coffee-break**
11:15-11:45	**Iv-3:** Innocente P. *Exploration of DTT Conventional and Advanced Divertor Configurations by Means of Edge Simulation Codes*
11:45-12:15	**Iv-4:** Ambrosino R. *Flexibility of the DTT Design*
12:15-13:00	**Discussion**
13:00-14:15	**Lunch**
13:00-15:00	**Poster Session 1**
Session 2: Liquid Metal PFC Chair: Hu J.S.	
15:00-15:45	**It-2:** Hu J.S. *Development of Liquid Lithium PFCs Providing an Alternative Design for DEMO Divertor*
15:45-16:15	**Iv-5:** Kolemen E. *Fast Flowing Liquid Metal Divertor Development at PPPL*
16:15-16:45	**Coffee-break**
16:45-17:15	**Iv-6:** Mazzitelli G. *Experiments with Liquid Metal Limiters on FTU*
17:15-18:00	**Discussion**
18:20-20:00	**Welcome Reception**

Tuesday, 14 November

Session 3: Divertor Physics and Modelling Chair: Wiesen S.	
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9:15-9:45	**Iv-7:** Fedorczak N. *Width of Turbulent SOL in Tokamaks: from Circular to Diverted Geometries*
9:45-10:15	**Iv-8:** Chen Y.P. *Simulations of SOL-Divertor Plasmas for Divertor Upgrade in EAST*
10:15-10:45	**Iv-9:** Brida D. *Power Exhaust and Detachment in Divertor Tokamaks with 3D Magnetic Perturbations – Experiments and Modelling in ASDEX Upgrade*
10:45-11:15	**Coffee-break**
11:15-11:45	**Iv-10:** Wischmeier M. *Impact of an Integrated Core/SOL Modelling on the Optimization of Tokamak Fusion Reactors*
11:45-12:15	**Iv-11:** Hoshino K. *JT-60SA Divertor Research Strategy and Radiative Scenario Modelling with Improving SONIC Integrated Code*
12:15-13:00	**Discussion**
13:00-14:15	**Lunch**
13:00-15:00	**Poster Session 2**
15:00-15:30	**Iv-12:** Asakura N. *Studies of Plasma Exhaust Scenario and Divertor Design for JA DEMO*
15:30-16:00	**Iv-13:** Canik J.M. *Predicting Highly Radiative Divertor Scenarios for Fusion Reactors*
16:00-16:30	**Coffee-break**
16:30-17:30	**Discussion**

Wednesday, 15 November

Session 4: PFC Development Chair: Visca E.	
8:30-9:15	**It-4:** Visca E. *Overview of Advanced Water-Cooled Divertor Target Design Concepts for European DEMO Reactor*
9:15-9:45	**Iv-14:** Greuner H. *Strategy and Results of High Heat Flux Testing of European DEMO Divertor Mock-Ups*
9:45-10:15	**Iv-15:** Neu R. *Tungsten Fibre-Reinforced Composites for Advanced Plasma Facing Components*
10:15-10:45	**Iv-16:** Mao Y. *Powder Metallurgically Produced Tungsten Fiber-Reinforced Tungsten Composites*

10:45-11:15	**Coffee-break**
11:15-12:00	**Discussion**
12:00-14:15	**Poster Session 3**
13:00-14:15	**Lunch**
Session 5: Effect of 3D Fields Chair: Liang Y.	
14:15-15:00	**It-5:** Liang Y. *Impacts of 3D Magnetic Perturbations on the Edge Plasma Transport and Stability on EAST*
15:00-15:30	**Iv-17:** Lee H. *Outer divertor Target Heat Fluxes During Resonant Magnetic Perturbation Induced ELM Suppressed Regimes in KSTAR*
15:30-16:00	**Iv-18:** Xiao B.J. *High-Confinement Steady-State Operation with Quasi-Snowflake Magnetic Configurations in EAST Tokamak*
16:00-16:20	**Coffee-break**
16:20-16:50	**Iv-19:** Kobayashi M. *Edge Radiation Control in Stochastic Magnetic Field and with RMP Application in LHD*
16:50-17:20	**Iv-20:** Masuzaki S. *Heat and Particle Transport from the Stochastic SOL to Divertor in the Large Helical Device*
17:20-18:05	**Discussion**
18:20-20:00	**Banquet Dinner**

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9:15-9:45	**Iv-21:** Wang L. *Particle and Power Exhaust for H-mode Operation over 100 Seconds with ITER-like Tungsten Divertor in EAST*
9:45-10.10	**Discussion**
10:10-11:30	**Summary Discussion** **Closing Remarks**
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Poster Sessions

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P-2	Rao B., *Conceptual Design of Magnetic Island Divertor in the J-TEXT tokamak*
P-3	Zhang X.D., *Fishtail Divertor - Swinging Strike Point For Active Control of Heat Load on Divertor Plant*
P-4	Ye M., *Simulation Study of Quasi-snowflake divertor with Ne seeding*
P-5	Poradzinski M., *Influence of the plasma parameters on the divertor power load for DEMO with tin divertor*
P-6	Xu W., *Active suppression of tungsten impurity influx using lithium aerosol injection in EAST*
P-7	Zuo G.Z., *Using Li to mitigate plasma-material interactions in EAST*
P-8	Chen D., *Halo current, disruption mitigation on EAST with tungsten divertor*
P-9	Wang Y., *Multi-Input Multi-Output control for EAST Quasi-Snowflake configuration*
P-10	Khanal R., *Self-consistent kinetic trajectory simulation model of magnetized plasma sheaths*

Poster Session 2	
P-11	Labit B., *Overview of Lower Single-Null Neutral Beam Heated ELMy H-Mode Plasmas in TCV*
P-12	Sang Ch., *Understanding the Role of Pumping in the Closed Divertor*
P-13	Canik J.M., *Pumping Optimization for the EAST Divertor Upgrade*
P-14	Varoutis S., *Numerical Investigation on the Design Space of a DEMO Divertor: A Quest for Achieving High Pumping Efficiency*
P-15	Mao R., *Plasma Simulation with SOLEDGE2D for Complex Divertor Configurations on HL2M*
P-16	Liu X.J., *Simulation Study of Divertor Detachment Density Threshold in EAST*
P-17	Mao Sh., *Radiative Divertor for CFETR by Impurity Seeding*
P-18	Kong D.F., *Simulation Study of Local Closure Effect of the Divertor Structure on Detachment Density in CFETR*
P-19	Xiang L., *Evolution of Impurities and Radiation Distribution in the edge to Boundary Region of Tokamak Plasma towards Divertor Detachment*
P-20	Chen J., *Comparison of Ar and Ne Radiative Divertor Plasmas in EAST*
P-21	Ding F., *Impact of Divertor Configuration on H-Mode Access and Confinement on EAST*
P-22	Wu K., *Radiative Feedback Control for Power Exhaust in EAST Long-Pulse Operations*

Poster Session 3	
P-23	Yao D.M., *EAST Lower Tungsten Divertor Concept and Technics Development*
P-24	Firdaouss M., *W Coatings on Divertor Secondary Areas*
P-25	Xu G.S., *Conceptual Design of the New Water-Cooled W/Cu Lower Divertor in the EAST Superconducting Tokamak*
P-26	Li Ch., *Theoretical Analysis on the Damages of Tungsten Plasma Facing Component under Both Steady-State Heat Loads and Transient Heat Fluxes in Fusion Devices*
P-27	Cao L., *Failure Analysis of EAST Lower Divertor*
P-28	Liu Ch., *Investigation on the Thermal Conductivity and Heat Flux Route of the Tungsten Divertor Monoblock under the Fracture Conditions for EAST Tokamak*
P-29	Liu P., *Fatigue and Fracture Analysis on ITER-Monoblock Heat Sink at Operating Temperature*
P-30	Xu T.J., *Status of Technical R&D Activity for EAST Lower Tungsten Divertor*
P-31	Zhou Z., *The Engineering Design of EAST Lower Advanced Divertor*
P-32	Dai Sh., *EMC3-EIRENE Modelling of 3D Impurity Transport in Argon Seeded EAST Discharges*
P-33	Shi Q., *3D Modelling of Tungsten Fuzz Growth under the Bombardment of Helium Plasma*
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Abstracts

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List of Posters:

It-1: Overview of SOL transport and detachment in alternative divertor geometries on TCV

C. Theiler[1], J.A. Boedo[2], B. Duval[1], N. Fedorczak[3], O. Février[1], A. Gallo[3], J. Harrison[4], E. Havlickova, B. Labit[1], B. Linehan[5], B. Lipschultz[6], R. Maurizio[1], B. Mumgaard[5], H. De Oliveira[1], H. Reimerdes[1], U. Sheikh[1], A.J. Thornton[4], C. Tsui[2,1], K. Verhaegh[6,1], N. Vianello[7], W.A.J. Vijvers[8], M. Wensing[1], and the TCV team[*] and the EUROfusion MST1 team[†]

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The flexibility of the TCV [1] magnetic geometry and its open divertor allow the exploration, in a single machine, of the whole zoo of alternative divertor geometries. In dedicated L-mode scans, divertor leg length, poloidal flux expansion, and target radius have been varied and additional X-points have been introduced near the core plasma and near the target [2,3].

Experiments in attached conditions reveal that the heat flux width λ_q, inferred from target infrared thermography and Langmuir probe measurements, increases with divertor leg length and is weakly dependent on poloidal flux expansion [4,5]. The effectiveness of divertor cross-field transport in reducing parallel heat fluxes at the target is found to decrease significantly with increasing poloidal flux expansion, as parameterized by the divertor spreading factor S [5]. The density threshold to detach the outer leg decreases with increasing divertor leg length, consistent with the observed increase in λ_q, but is otherwise fairly insensitive to geometrical modifications such as poloidal flux expansion and target radius [2,3].

New spectroscopic techniques [6] reveal that both a reduction of ionization source due to power starvation and volume recombination can contribute to the Isat-rollover, with the former generally dominating at the rollover-onset. Radiation levels in the divertor leg increase by 50-100% when increasing divertor volume in the poloidal flux expansion and the leg length scans. The movement of the radiation peak from the target to the X-point as a function of density is slowed down significantly with increasing poloidal flux expansion, improving detachment controllability. Yet, in all the geometries, power is primarily radiated from the X-point region for densities above the detachment threshold [2]. Introducing an additional X-point in this region is found to be an efficient tool to control the location of this X-point radiator and to shift it outside the confined region of the plasma [3].

Results from first experiments aiming to extend these studies to H-mode operation will be presented and future capabilities with in-vessel neutral baffling [7,8] and the resulting changes predicted by SOLPS simulations will be discussed.

References:

[1] S. Coda *et al.*, Nucl. Fusion **57**, 102011 (2017)
[2] C. Theiler *et al.*, Nucl. Fusion **57**, 072008 (2017)
[3] H. Reimerdes *et al.*, Nucl. Fusion 2017, in press
[4] A. Gallo *et al.*, Plasma Phys. Control. Fusion 2017, in press

[*] See the author list of Coda et al., Nucl. Fusion 57 (2017) 102011
[†] See the author list of H. Meyer et al., Nucl. Fusion 57 (2017) 102014

[5] R. Maurizio *et al.*, Nucl. Fusion, submitted
[6] K. Verhaegh *et al.*, NME 2017, in press
[7] A. Fasoli *et al.*, Nucl. Fusion **55**, 043006 (2015)
[8] H. Reimerdes *et al.*, NME 2017, in press

It-2: Development of liquid lithium PFCs providing an alternative design for DEMO divertor

J.S. Hu[1,2], G.Z. Zuo[1], R. Maingi[3], D. Andruczyk[4], D.N. Ruzic[4], Z. Sun[1], W. Xu[1],Q.X. Yang[1], M. Huang[1], X.C. Meng[5], X.Z. Gong[1], B.N. Wan[1], J.G. Li[1]

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We present here about investigation on the development of the design of liquid lithium (Li) limiters and researches of the interaction between plasma and the limiter, aiming providing an alternative resolution for the divertor design of future DEMO device. From laboratory test to experiments in HT-7 and EAST tokamaks, we have successively designed and tested static liquid Li limiter respectively with a free-surface and a capillary porous structures (CPS), and flowing liquid Li limiter (FLiLi) respectively applying thermoelectric magnetohydrodynamic (TEMHD) effect and a thin flowing film concept [1,2]. Meanwhile, a few key techniques were developed, such as to drive Li flowing, to improve the wetting and flowing of Li, to control Li erosion, and also how to improve the heat flux exhaust, etc..In addition, we found that Li limiters with a CPS and thin flowing film concept were beneficial for avoiding strong Li emission and then to decrease disruptive plasmas[3]. Specially, FLiLi with the concept of thin flowing film was controllably driven by an in-vessel DC electromagnetic (EM) pump, using the toroidal magnetic field of EAST[4]. Recently, an upgraded second-generation FLiLi using a few new techniques improve the liquid lithium coverage uniformity to >80%, on stainless steel plate. Exciting positive results, such as the improvement of the performances of H-mode plasmas, controllable Li emission for the reduction of divertor heat flux [5] and the mitigation of ELMs, were found. Expecting to improvement of wetting and flowing of Li and base material, a new FLiLi using Mo instead of SS as support surface for Li is being developed and will be tested in HIDRA in Illinois and EAST at the end of 2017. Those efforts would provide reference to design divertor with high heat flux in future reactors by allowing for a self-healing, self-replenishing surface with no susceptibility to neutron damage to partly ameliorate lifetime and power-exhaust issues of PFCs.

References:

[1] G. Z. Zuo, et al., J. Nucl. Mater. **415**, S1062 (2011).
[2] J. S. Hu, et al., Fusion Eng. Des. **89**, 2875 (2014).
[3] G. Z. Zuo, et al., Fusion Eng. Des. **89**, 2845 (2014).
[4] J. S. Hu, et al., Nucl. Fusion **56**, 046011 (2016).
[5] G. Z. Zuo, et al., Nucl. Fusion **57**, 046017 (2017).

It-3: Modelling radiative power exhaust in view of DEMO relevant scenarios

S. Wiesen[1], M. Wischmeier[2], M. Bernert[2], M. Groth[3], A. Kallenbach[2], R. Wenninger[2], S. Brezinsek[1], F. Reimold[1], M. Beckers[1], W. Biel[1]

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Power exhaust in future electricity producing fusion device features difficulties which, compared to ITER, are harder to tackle in terms of operational control requiring an improved physics understanding. Primarily, the heat flux problem must be solved in order to allow plasma facing components (PFCs) to withstand a maximum heat flux not exceeding $5MW/m^2$ in steady state on timescales lasting days or weeks. For a given P_{loss}/R, with P_{loss} being the power arriving from the core into the pedestal region, a total dissipated fraction $f_{diss}=P_{diss}/P_{loss}$ close to 95% or higher is required to reduce the total target heat load q_t. P_{diss} covers the diverted plasma edge region, the full scrape-off layer (SOL) and parts of the confined core region and thus accounts for losses from (impurity) radiation, transport towards the main-chamber and charge-exchange collisions with neutrals. A pronounced or even completely detached divertor regime is to be exploited to allow for the required pressure loss along the field that induces a roll-over of the target particle flux Γ_t. Consequently, the heat flow towards the target is mitigated and the deposited energy of recombining particles at the surface is minimized. Low divertor temperatures $T_{e,div}$ < 1-2 eV are additionally required to keep the total t arget heat loads and erosion yields within material limits inducing also recombination of particles within the divertor volume. In such low-T_e/high-n_e divertor regime strong interaction with neutral particles (D, T, He-ash and impurities) is inevitable. With the transport of neutral particles disconnected from the magnetic field a highly non-linear regime evolves which makes a coupled plasma-fluid/neutral-kinetic transport model approach indispensable. Validated 2D/3D numerical tools like SOLPS-ITER, EDGE2D-EIRENE or EMC3-EIRENE are required to assess the potential operational regimes in varying geometries in a predictive way (single-null, double-null, snowflake, etc.). For extrapolation to DEMO the chosen numerical plasma-edge model package is required to be scalable, too. The numerical analysis of similarity studies on power exhaust is a task which is currently being established between JET and AUG devices that should assist in this respect.

Secondly, the response of the SOL onto the core region needs to be taken into account. Recently, the so-called high-field side high-density (HFSHD) region has been identified in various tokamaks. The HFSHD region is formed by a combination of high recycling at the inner target and power crossing the separatrix allowing ionization of recycled neutrals. The main impact of the HFSHD is on the pedestal fueling pattern with the effect of a shift of the density profile due a change of separatrix density and hence, a quantitative change in pressure driven pedestal transport is a consequence. The interlink with the significant radiation loss in the same region must lead to additional constraints for performance relevant parameters as for example top pedestal density (and thus fusion product) as well as the distance to the H-to-L transition threshold. The understanding of the HFSHD region has progressed recently by modelling. However the impact on the pedestal fueling (that may also include transient effects from ELMs in H-mode) can only be tackled rigorously by using integrated models that merge the different transport regions together, i.e. plasma core, SOL/divertor and plasma-wall interaction (PWI).

Thirdly, from the engineering point of view reduced models are required to design the operational and control parameters of a future device design like DEMO. The pathway to control power and particle exhaust without losing main plasma performance in a future device is still hampered by unknowns in the underlying plasma physics effects. To what extent the models for the complex interplay of core/SOL/PWI can be reasonably reduced is a current field of interest. Fundamental for a plausible model reduction is the existence of databases that map out the operational parameters in existing devices (size, power, density, impurity concentrations, etc.), complemented by a series of numerical simulations.

This paper portrays the state-of-the-art of modelling capabilities within the fusion community. An update on the important validation process of numerical tools is presented. Missing model features that are required for a credible DEMO numerical model are highlighted as well as the gaps in physics understanding are discussed.

It-4: Overview of advanced water-cooled divertor target design concepts for European DEMO reactor

Eliseo Visca[1], J.H. You[3], F. Crescenzi[1], H. Greuner[3], T. Barrett[4], M. Richou[2], A.V. Müller[3]

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One of the most critical scientific and technological challenges for a nuclear fusion reactor, as defined for the European-DEMO reactor, is the capability to handle high power exhaust. The divertor is the key in-vessel component being in charge of power exhaust and removal of impurity particles. In particular, DEMO divertor targets have to withstand extreme thermal loads with local peak heat flux reaching up to 20 MW/m^2 during slow transient events. A sufficient heat removal capability of divertor targets against normal and transient operational scenarios is then required where a cumulative neutron dose of up to 18 dpa has to be considered for the structural material. To find a feasible technological solution an integrated R&D program for the European DEMO reactor was launched in the framework of the EUROfusion Work package "Divertor" (WPDIV). Pre-conceptual design activities as well as technological verification of the design concepts by means of manufacturing and high-heat-flux testing of target mock-ups are in progress.

Currently, besides an ITER-type monoblock design being considered as baseline model, seven different kinds of design concepts are being developed in WPDIV as water-cooled target for the European DEMO. They differ from each other either by materials or geometry. While pure tungsten is considered as reference armour material for all concepts, application of novel copper-based heat sink materials is one R&D line (e.g. copper-tungsten composites reinforced with particles or wires and laminated tungsten). Enhancement of joining quality by novel interlayer material (graded coating) between heat sink and armour is another concept. Mitigation of the heat flux concentration at the cooling tube wall is an alternative innovative concept ('so called thermal break' concept) where a thermal barrier interlayer is inserted between the W monoblock and the Cu alloy tube.

The currently investigated design concepts are presented together with the latest heat flux testing results carried out in GLADIS facility.

It-5: Impacts of 3D magnetic perturbations on the edge plasma transport and stability on EAST

Y. Liang[1,2,3], Y. Sun[2], L. Wang[2], L. Zeng[2], N.C. Wang[3], H.X. Zhang[2], M.N. Jia[2], W. Feng[2], T. Zhang[2], X. Han[1,2], Y. Liu[2], M. Dostal[1], P. Drews[1], Y. Gao[1], E. Wang[1,2], Y.L. Li[1,2], J.P. Qian[2], H.D. Xu[2], M. Cao[2], Z.B.Song[3], Q. Zang[2], H.H. Wang[2], and the EAST-team*

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The next generation fusion machines, like ITER and DEMO, will need a reliable method for controlling or suppressing large edge-localized modes (ELMs) [1]. Resonant magnetic perturbations (RMPs), which change the magnetic topology of the confined plasma, have been applied to completely suppress ELMs or to mitigate ELMs in several devices [2,3,4,5]. Although the physics mechanism is still unclear, experimental results from those experiments demonstrate that the magnetic topology plays a key role in plasma transport, edge magnetohydrodynamic stability, and interactions between the plasma and the first wall, particularly with the divertor [5,6,7].

On EAST, ELM suppression/mitigation has been achieved using low *n* magnetic perturbations induced by either a set of in-vessel coils [5] or the lower hybrid wave (LHW) [7] in low rotating H-mode plasma with RF heating. During the transition between ELM mitigation and suppression, a clear nonlinear variation of the plasma response to low *n* magnetic perturbations was observed. This nonlinear variation of the plasma response indicates a change of magnetic topology occurs at the plasma edge. Recent results from heat propagation experiment on EAST show significant impacts of 3D magnetic perturbations on the edge transport and the heat distributions on the diverter plates. In this paper, the role of magnetic topology on the edge plasma heat and impurity transports, ELM stability, and its impacts on the PWI will be discussed.

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It-6: Compatibility of dissipative divertor operation with core operational scenarios

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The compatibility of dissipative, detached divertor operation with high confinement core plasma scenarios is reviewed. The divertor and SOL plasma in future DEMO-scale tokamaks must simultaneously satisfy boundary conditions at the divertor target and at the upstream separatrix interface to the core plasma. While the requirements for the divertor target plasma to avoid excessive erosion and melting are well established, the upstream separatrix requirements are less well defined. The upstream parameters separatrix electron and impurity densities are particularly important for designing and assessing the viability of future divertor solutions. In a DEMO-scale tokamak fueling of the core plasma from divertor recycled neutrals will be greatly reduced offering the potential of greater flexibility in separately specifying the core and divertor densities. In DIII-D the ratio of separatrix to pedestal density, $n_{e,sep}/n_{e,ped}$, has been observed to vary from $\leq$30% to $\geq$60%, though this ratio is still dominated divertor recycling. Such a range for $n_{e,sep}$ in a DEMO-scale tokamak might even negate the need for alternative divertor configurations. However, it is unknown whether a high performance H-mode pedestal is achievable with a very high ratio of temperature to density gradient in the pedestal. Similar uncertainties exist for requirements on the upstream separatrix impurity density. While seeded impurities are necessary for dissipating exhaust power in the divertor, the impurity density must not cause the core plasma to exceed the limits for fuel dilution and radiation. If high ratios of pedestal temperature to density gradients are indeed achievable then greater levels of divertor impurity density may be tolerable in DEMO-scale tokamaks than in existing devices. These and other issues core and divertor plasma compatibility will be discussed with the aim of highlighting high priority areas for future research.

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Iv-1: Simulations of Passively Stable, Fully-Detached Regimes in Long-Legged Divertor Configurations

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Passively-stable fully detached divertor regimes have been found in numerical modeling of divertor configurations with radially or vertically extended, tightly baffled, outer divertor legs, with or without a secondary X-point in the leg volume, for a broad range of input power from the core [1]. Here a comparative computational study of detached divertor operation is presented, for a variety of divertor configurations and model assumptions, expanding on the initial work reported in Ref. [1]. The parameters are based on those of the ADX tokamak design [2], and the simulations are carried out with the tokamak edge transport code UEDGE [3]. The simulations show that long-legged divertors have a large increase of the peak power handling capability, by up to an order of magnitude, compared to conventional divertors. The key physics for the detached divertor regime in these simulations combines interplay of strong convective plasma transport to the outer wall, confinement of neutral gas in the divertor volume, geometric effects including secondary X-point, and atomic radiation. As the power from the core is varied, the detachment front merely shifts up or down in the leg but remains stable. The present work addresses sensitivity of the detached divertor regime to various parameters used in the model, including the anomalous plasma transport, neutral transport, impurity radiation, and geometry of plasma-facing material surfaces.

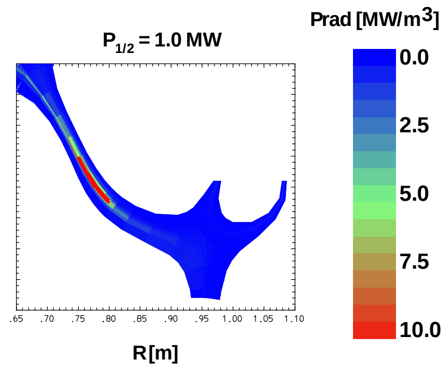

Fig.1. Radiation emissivity in the outer divertor leg of ADX for an X-point target geometry. The radiation front location in the leg is passively stable, shifting up or down the leg depending on the input power.

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Iv-2: Taming Plasma-Materials Interface for Steady-State Fusion

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It poses increased challenge to develop a viable plasma-materials interface solution for next-step long pulse tokamaks, such as a Fusion Nuclear Science Facility (FNSF), which will have lower plasma density than ITER for high performance steady-state current drive and high duty cycle operation. An innovative divertor concept, named small angle slot (SAS), has been developed on the DIII-D tokamak facility at General Atomics to address the challenge of efficient divertor heat dispersal compatible with non-inductive current drive in future tokamaks. The goal of this advanced divertor solution is to optimize the geometry of the target and baffle for the control of neutrals to provide the most efficient and complete energy deposition possible. Modeling using a multidimensional boundary plasma-fluid and Monte-Carlo-neutral transport code package (SOLPS) shows that SAS leverages the strong synergistic effects of a critical small target angle and a gas-tight slot geometry to achieve strongly dissipative divertor plasmas over the entire divertor surface at relatively low main plasma densities. This provides a potential means for simultaneous control of heat flux and erosion at the material surface, which is mandatory for high-performance steady-state fusion plasmas. Initial tests on DIII-D have demonstrated these promising features predicted by the models for a range of plasma configurations and conditions. Efforts will also be made on DIII-D to evaluate impact of high-Z target plasma facing materials (PFM) in advanced divertors, and extend PMI evaluation from the divertor to main chamber to develop an integrated core-edge solution for next-step steady-state fusion devices. Reactor studies have silicon carbide (SiC), along with W, as a PFM of choice, but limited tests have been performed in realistic fusion environments. We have developed a strategic plan on DIII-D to validate SiC as a PFM candidate for a fusion reactor, as well as to evaluate advanced high-Z PFMs in collaboration with the broad materials development community during the next five-year period.

This work was performed in part under the auspices of the U.S. Department of Energy by General Atomics under DE-FC02-04ER54698.

Iv-3: Exploration of DTT conventional and advanced divertor configurations by means of edge simulation codes

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The development of a reliable solution for the power and particle exhaust in a reactor, in order to ensure acceptable conditions at the walls while maintaining sufficient core performance, is recognised as one of the major research challenges towards the realisation of a fusion power plant [1]. In order to mitigate the risk that the conventional divertor solution that will be tested in ITER may not extrapolate to DEMO, alternatives must be developed. The role of the DTT facility is to bridge the gap between today's proof-of-principle experiments and DEMO [2]. DTT has been designed to study a large suite of alternative magnetic configurations, including flux flaring towards the target (X divertor), increasing the outer target radius (Super-X) and movement of a secondary x-point inside the vessel (X-point target) as well as the entire range of Snowflake (SF) configurations [3]. Here, first comparative edge studies of conventional Single Null (SN) and alternative configurations by using EDGE2D-EIRENE [4], TECXY [5] and SOLEDGE2D-EIRENE [6] codes will be presented. A closed divertor, with a full W wall, no impurity seeding and a high level of power crossing the separatrix $P_{SOL}\approx 40$MW, have been considered in the simulations. In addition, the transport coefficient has been set up constant and an outer midplane decay length of 3 mm has been assumed. A density scan for both the conventional and advanced configuration has been performed in order to investigate the behaviour of the different magnetic divertor solutions. In the conventional scenario high power loads are foreseen by the code independently from the density, with a peak values higher than 20 MW/m^2. A similar behaviour is also observed in the case of partially detached conditions where the strike point electron temperature falls below 5 eV leading to a clear indication of the roll-over of the density and of the saturation current. However, this condition is reached for upstream density higher than the one foreseen even in the high density scenario. On the contrary, in case of a SF like configurations, manageable values of the power load are obtained also for the medium density scenario. Furthermore, the codes predict detachment conditions for lower value of the upstream density. This behaviour is probably related to the benefit deriving from the geometrical feature of the SF like configurations in terms of increase in the flux expansion and connection length.

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Iv-4: Flexibility of the DTT design

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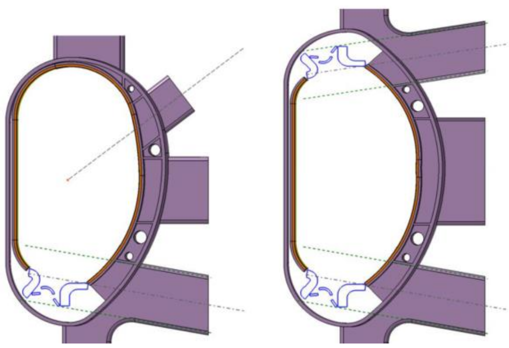
One of the conclusions of the DTT workshop held in Frascati on 19-20 June 2017 was that the design of a Divertor Tokamak Test facility [1-3] should offer sufficient flexibility to be able to incorporate the best candidate divertor concept (e.g. conventional, snowflake, super-X, double null, liquid limiter) even at a later stage of its realization.

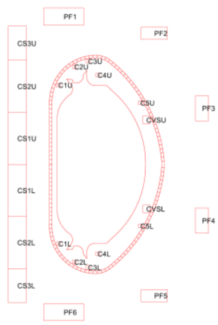
In this respect, four main points have been risen concerning the present status of the proposal and the possible directions to be explored: i) flexibility of the machine, as stated above; ii) up-down symmetry, so as to properly test double null configurations; iii) minimum amount of the additional power coupled to the plasma needed to guarantee significant results in view of DEMO; iv) the best additional power mix (coming from the various sources: ICRH, ECRH and NBI) in view of the flexibility requirements.

Here we explore the possibility to define an up-down symmetric basic machine by reflecting the lower part of the DTT tokamak with respect to the equatorial plane (superconducting TF coils, superconducting PF coils, in vessel coils, first wall, divertor, vacuum vessel and ports). Figure 1 shows the vacuum vessel and the PF system.

Plasma available volume increases in the up-down symmetric version of DTT. This makes it possible to increase the reference values of the plasma current. At the same time, however, this might have an impact on the costs, for which a slight revision of the main parameters could be considered in the future, e.g. a 5% reduction of the major radius.



a)



b)

Fig. 1: a) First wall, divertor, vacuum vessel and ports in the original DTT proposal; b) same for the up-down symmetric machine; c) PF system of the up-down symmetric machine.

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Iv-5: Fast flowing liquid metal divertor development at PPPL

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We present the results and developments from Liquid Metal eXperiment, LMX, and Flowing LIquid metal Torus, FLIT, at Princeton Plasma Physics Laboratory, PPPL.

Currently, there no solid material that can handle both the neutrons and the extreme heat flux that will be present in a fusion reactor. A fast flow liquid metal divertor is an alternative that aims to handle all the cooling requirements at the divertor allowing a steel alloy surface behind to be designed only for the neutron damage. Using lithium as the liquid metal may also have the added advantage that hydrogen isotope and possibly helium pumping in addition to core confinement enhancement. This concept has no limit on the heat that can be handles only an increase in liquid metal velocity is required. The main issue to overcome in this concept is the stabilizing the fast flow under MHD effects.

LMX at PPPL is used to study the fast liquid metal in channel configuration with magnetic field up to 0.3 Tesla and 2 m/s flow speeds. FLIT is an upcoming torus device at PPPL is designed to look at the annular flow at up to 1 Tesla and 10 m/s. We present the results from LMX and the design and construction of FLIT.

At LMX we studied the heat transfer in liquid metal, and 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 shapes such as delta-wing shapes and various dimple configurations at the bottom of the channel and their effects in the experiment and comparison with the numerical simulations are shown. The effect of currents running through the liquid metal was explored. This effect is important for magnetic propulsion and slow stabilization. We developed an analytical and numerical model for the Lorentz forces. The relationship between the Lorentz force flow parameters and the hydraulic jump location is shown.

FLIT focuses on a liquid metal divertor system suitable for implementation and testing in present-day fusion systems, such as NSTX-U. It is designed as a proof-of-concept fast-flowing liquid metal divertor that can handle heat flux of 10 MW/m^2 without an additional cooling system. The 72 cm wide by 107 cm tall torus system consisting of 12 rectangular coils that give 1 Tesla magnetic field in the center and it can operate for greater than 10 seconds at this field. Initially, 30 gallons Galinstan (Ga-In-Sn) will be recirculated using 6 jxB pumps and flow velocities of up to 10 m/s will be achieved on the fully annular divertor plate. FLIT is designed as a flexible machine that will allow experimental testing of various liquid metal injection techniques, study of flow instabilities, and their control in order to prove the feasibility of liquid metal divertor concept for fusion reactors. Details of the design of FLIT will be presented.

Iv-6: Experiments with liquid metal limiters on FTU

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The problem of power exhaust is one of the major issues to solve for a future reactor. In the last decade liquid metals as plasma facing components (PFC) have been proposed as possible alternatives to solid materials. The use of liquid metals can improve the lifetime and reliability of PFCs because they are not subjected to erosion with dust formation and to a deterioration of their thermo mechanical properties due to intense hydrogen and helium influx and neutron irradiation as for solid materials.

Since 2006 on FTU liquid metal limiters were exposed to plasma discharges. A capillary porous system was adopted to confine the liquid metal to avoid droplet formation. At the beginning we have used Li as liquid metal but in the last year experiments with tin were performed, too. The main goals of the experimental campaigns were to determine the operational window for both materials and the compatibility with plasma performance especially for tin (Z=50).

The liquid metal limiter is inserted from one vertical port at the bottom of the machine and its radial position can be varied shot by shot from 3.5 cm outside the last closed magnetic surface (LCMS) up to 2 cm inside the LCMS as defined by the TZM toroidal limiter acting as main limiter.

The liquid limiters can be cooled but here, as a first step, we will show results obtained without the cooling system active. Heat loads less than 10 MW/m^2 for Li and 18 MW/m^2 for tin were withstood by the liquid limiters without any damage. Strong evaporation is present at this power levels but nevertheless no degradation in plasma performance has been observed. For both Li and Tin plasma pollution is limited and compatible with reactor operation.

The implications of these results for the operations of a future reactor will be discussed.

* Overview of the FTU results, G.Pucella et al., Nucl. Fusion 57 (2017) 102004

Iv-7: Width of turbulent SOL in tokamaks: from circular to diverted geometries

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Control of power loads on plasma facing components in current and future tokamak reactors largely depends on perpendicular transport properties setting the scrape of layer width. This contribution aims at showing that interchange turbulence is a likely candidate, first in inner limited circular geometry, but also in diverted one. After confronting experimental evidences and simulation results, implications for future reactors are discussed.

For limited circular geometries, a 2D interchange transport model is validated against a broad set of fluctuation properties and mean density and heat load profiles collected in Tore Supra [1]. In particular (1) the ExB drift velocity of plasma filaments (blobs) are reproduced by an isolated blob model, and (2) values of the SOL width are matched by a regression scaling - on global control parameters - constructed from numerical simulations. Both agreements are within 30% error. The model scaling for λ_q depends mainly on the poloidal magnetic field strength, including weaker but finite sensitivity with machine size and total magnetic field strength. Predictions for ITER start-up phase fall in the range from multi-machine regressions [2], albeit ignoring the possible existence of a narrow feature.

Application of the regression scaling to diverted geometry has to suffer the evidence that λ_q is generally much smaller in this configuration than in circular limited plasmas, which cannot be explained by the previous model in its state. On the other hand:

- Experimental scaling laws constructed on JET and AUG L-mode lower single null data [3] return a parametric sensitivity of λ_q in good agreement with the model.
- Recent TCV data [4] confirm this agreement. It points toward the existence of a positive sensitivity of λ_q with machine major radius.
- Recent 3D flux driven turbulent simulations of the edge of both circular and diverted plasmas, made with TOKAM3X, show that (1) interchange turbulence dominates transport in both cases, (2) λ_q is much narrower in diverted than inner limited configurations. Ongoing works point toward the role of magnetic shear and expansion effects on transport mitigation.

Implications are twofold. First, recent estimates of λ_q for ITER, either based on multi-machines extrapolations or drift-based models, could be too small by a factor of at least 2. Second, interchange turbulence is also likely to take place in the divertor volume. Simulations with TOKAM3X on equilibria with increasing length of the outer divertor leg - in TCV-like geometry - show that λ_q increases with the leg length whereas divertor spreading S does not [5]. These results are in good agreement with recent experimental evidences from TCV [4,5]. Beyond questioning the physical interpretation of λ_q & S, it opens new perspectives in the optimization of turbulent transport in alternative divertor configurations.

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Iv-8: Simulations of SOL-Divertor Plasmas for Divertor Upgrade in EAST

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In order to adapt to the higher heating power and larger heat fluxes on the targets, increase the capacity of its erosion-resistance and contribute to the study of ITER-like divertor, the lower divertor on EAST will be upgraded by using the W target plates and optimized divertor geometry according to R&D of EAST divertor. Low power onto the divertor target and sufficient divertor particle removal is the key driver for the divertor upgrade. Based on the simulations and physics design, a new divertor structure, structure 6B, is proposed by the physics design team. The new structure includes the vertical inner target, dome, horizontal outer target and baffle. The new lower divertor with 6B structure is simulated and analyzed by using SOLPS-ITER, aiming at the assessment of the computation physics design. The simulations deal with following key issues and main simulation results are drawn.

1. Operation regimes by density scan

In order to survey the divertor operation over a wide range of plasma parameters and find the operating regimes for the divertor plasmas, the separatrix density n_{esep} is varied extensively, while the total plasma heat flux Pt to the computational region through the core-SOL interface is fixed at Pt=4.0 MW which is similar to the heating power used in the present EAST experimental campaigns. Conduction limited (high recycling) operation regime appears at inner target with separatrix density $n_{esep}=2.0\text{-}2.5\times10^{19}m^{-3}$ and at outer target with $n_{esep}=3.0\text{-}4.0\times10^{19}m^{-3}$. With the density increase the inner target will enter detachment operation earlier than the outer target.

2. Heat flux

The actual power and particle fluxes deposited on the target plate of divertor are determined not only by the parallel fluxes, but also by the angle of incidence of the field line onto the target plate itself. What we are concerned about most is the heat flux at the horizontal outer target with larger incident angle of the field line. The peak heat fluxes have been obtained by density scan. Maximal peak heat flux $P=13MW/m^2$ with $n_{esep}=2.0\times10^{19}m^{-3}$ at outer target and $P=4.7MW/m^2$ with $n_{esep}=1.0\times10^{19}m^{-3}$ at inner target happen in the sheath limited (low recycling) operation regime.

3. Particle exhaust

Particle exhaust is enhanced by using three slots, i.e., slot 1 between the inner target and dome, slot 2 between dome and outer target and slot 3 between outer target and baffle. The simulation results can show pumping particle flux through each slot. $s1=6.34\times10^{20}s^{-1}$, $s2=3.04\times10^{20}s^{-1}$ and $s3=4.45\times10^{20}s^{-1}$ at $n_{esep}=1.0\times10^{19}m^{-3}$ and increases respectively to $s1=5.98\times10^{21}s^{-1}$, $s2=6.00\times10^{20}s^{-1}$ and $s3=4.23\times10^{21}s^{-1}$ at $n_{esep}=4.0\times10^{19}m^{-3}$. s1, s2 and s3 are important parameters for the control of particle exhaust.

Iv-9: Power Exhaust and Detachment in Divertor Tokamaks with 3D Magnetic Perturbations – Experiments and Modelling in ASDEX Upgrade

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A key challenge for future fusion devices, such as ITER and DEMO, is the control of power exhausted from the main plasma. If the steady state peak heat flux onto the divertor targets is not mitigated, it will exceed the material limit of 5-10 MW m^{-2} substantially. The strategies to reduce the target heat flux and reach a detached divertor have been investigated in numerous experimental and numerical studies. These studies largely assume an axisymmetric magnetic field geometry. However, in recent years 3D Magnetic Perturbations (MPs) have been increasingly used in several tokamaks, in order to mitigate or even suppress edge localized modes. MPs break the toroidal axisymmetry and lead in effect to a change in the radial transport. If they find application in future divertor tokamaks, their impact on power exhaust and the divertor target heat flux needs to be understood.

MPs might have beneficial as well as harmful consequences. Under attached conditions, IR thermography shows the occurrence of toroidally localized heat flux maxima. It has been speculated that these may 'burn-through' an elsewhere detached divertor leading to intolerable heat flux densities. If this problem arises in ITER, it would be necessary to apply countermeasures, such as rotating the MP field, which would entail substantial engineering efforts. On the other hand, MPs could lead to the beneficial effect of an increased toroidally averaged power decay length $< \lambda_q >$. This effect would not only lead directly to a decreased target peak heat flux due to an increased plasma wetted area, but also increase the volume in which impurities, such as nitrogen, can radiate efficiently.

In this contribution, experimental and numerical results of ASDEX Upgrade (AUG) discharges with MPs will be presented and their implications in view of ITER discussed. Based on measurements by divertor Langmuir probes it will be argued that a burn-through event is unlikely in ITER, since toroidal asymmetries are smoothed out by diffusive transport at low divertor plasma temperatures. In addition to that, the comparison of the experiments with simulations by the transport code EMC3-EIRENE revealed several strong indications for the existence of screening currents induced in the plasma that reduce the amplitude of the MP fields in the confinement region. Due to this the impact on both, the toroidally averaged fall-off length $< \lambda_q >$ and the radiation is rather small and barely measurable with the present diagnostics in AUG. In ITER, however, where $< \lambda_q >$ is predicted to be of the same order, while the radial perturbation of field line paths is much larger, a substantially stronger effect is predicted.

[*] See the author list "H. Meyer et al 2017 Nucl. Fusion **57** 102014"

Iv-10: Impact of an integrated core/SOL modelling on the optimization of tokamak fusion reactors

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A 0D divertor and scrape-off layer (SOL) model [1,2,3], which provides the heat flux and the temperature at the plates for given upstream conditions and seeded impurity concentrations, has been coupled to the 1.5D core transport code ASTRA [4, 5]. The resulting numerical tool has been employed for various parameter scans in order to identify the most convenient choices for the operation of electricity producing tokamak devices. The main result we found is that the curves at constant electrical power output are closed on themselves in the R B_T plane, i.e. no improvement would be achieved by further increasing R or B_T once the maximum has been reached. In particular, for high R and B_T , the need of auxiliary power to compensate the large synchrotron radiation losses from the main plasma is such to overcome any increase in the generated fusion power. Viceversa, at low B_T and large R, the necessary impurity concentrations for the divertor compatibility lead to a significant dilution of the main plasma, thus compromising the total fusion outcome - unless unrealistically high divertor compression factors are assumed. Also, some preliminary considerations about the presence of a hysteresis on the required impurity concentration for divertor detachment are reported.

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Iv-11: JT-60SA divertor research strategy and radiative scenario modelling with improving SONIC integrated code

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Japanese and European fusion research communities have developed the JT-60SA research plan [1], which summarizes strategies for JT-60SA, a superconducting large tokamak, for addressing issues for ITER and DEMO. In order to explore any possibilities of DEMO operational regime, JT-60SA will drive fusion research with a wide spectrum. For this, flexible upgrade plans for the hardware such as heating systems and divertor plasma facing components are considered. In addition, the research plan will enhance contribution to the studies for ITER support and DEMO steady-state operation, considering the latest ITER timeline.

One of the missions of JT-60SA is to sustain steady-state high-β plasmas. In such plasmas, significant heat load onto the divertor plates is expected. For mitigating the heat load, highly-radiative detached plasma is promising. In order to access high-recycling and then detached operational regime even with low-density high-β plasmas, JT-60SA will employ a V-shaped corner at the outer divertor such as in ITER, which can enrich neutral density. Further, the divertor plasma performance is studied by SONIC, an integrated edge plasma transport code. One of the features of SONIC is treatment of the impurity transport by a kinetic model, which has advantages in the transport modeling, such as the thermal force, plasma wall interactions etc. The SONIC simulation shows that impurity seeding is needed to reduce the peak heat load below 10 MW/m^2 for carbon divertor plate. In the analysis, SONIC needs to simulate another impurity besides intrinsic impurity from wall if one is seeded. All previous SONIC simulations ([2], [3] etc.) have treated one impurity with a kinetic model and the other with a modified coronal model, which does not simulate transport but considers only radiation.

In this study, we have developed a new integrated modelling framework using MPI data exchange. By introducing the new framework, two impurities and more, C and Ar for instance, can be treated simultaneously with a kinetic model. The radiative divertor plasma scenario is re-assessed by the improved SONIC code. As the same as the previous conclusion, the new SONIC simulation shows that radiative divertor plasma scenario compatible with the low-density high-β plasma can be achieved. It is also shown that the required gas puffing rate both for D_2 and Ar are decreased due to the carbon sputtering at the attached region and its transport.

After the primary missions of JT-60SA are accomplished, the carbon component will be replaced by the tungsten coated carbon (~2030). JT-60SA explores compatibility of steady-state high-β plasmas and the full tungsten wall. Since the kinetic effects become further important for tungsten, the radiative scenario modeling will be studied by SONIC using a kinetic impurity transport model for multi-impurity species.

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Iv-12: Studies of Plasma Exhaust Scenario and Divertor Design for JA DEMO

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The power exhaust scenario for the feasible DEMO plasmas and the appropriate divertor design have been studied with a high priority in the steady-state Japan (JA) DEMO with the fusion power of 1.5 GW-level and the major radius of 8 m-class [1]. At the same time, concept of the heat removal components based on the ITER technology, arrangement of the coolant routing, and the cassette body were integrated for the divertor design. The radiative cooling by the impurity seeding is a primary approach for the power exhaust, and the total radiation power fraction in the main plasma, SOL and divertor (P_{rad}/P_{heat}) is increased in order to reduce the peak heat load (q_{target}) lower than 10 MWm^{-2} level. The divertor design appropriate for *high* P_{sep}/R_p (~30 MWm^{-1}) is required, provided that f_{rad}^{main} (= P_{rad}^{main}/P_{heat}) is slightly larger than ITER-level (0.40-0.45) with Ar impurity fraction of n_{Ar}/n_e = 0.6-0.7%. A divertor design based on the ITER divertor with extending the divertor leg (L_{div} =1.6m) was proposed, and the plasma performance has been studied by divertor simulation code (SONIC). Operation window of P_{sep}/R_p (P_{sep} = 250-300 MW) and radiation fraction in the SOL and divertor ($f_{rad}^{div+SOL}$ = $P_{rad}^{div+SOL}/P_{heat}$) determined by Ar transport are determined for the acceptable q_{target} (< 10MWm^{-2}), and the characteristics of the detachment plasma depending on some important physics parameters such as edge density and plasma diffusivity are discussed.

An integrated design concept of the water-cooling heat sink, i.e. W-monoblocks with Cu-alloy (CuCrZr) and Reduced Activation Ferritic Martensitic steel (F82H) pipes, for the long leg divertor was proposed. The replacement of the heat sink (W-monoblock target unit with Cu-alloy pipe) due to higher neutron irradiation and material displacements per atom rate (DPA) is an important issue for the steady-state DEMO divertor design. The Cu-alloy pipe is applicable to handle the high heat flux near the strike-point, where DPA is estimated to be 0.5-1.5 per year by neutronics calculation. An arrangement of the coolant routing with the different temperature and pressure water conditions was investigated. The heat transport analysis of the W-monoblock and Cu-alloy pipe under the peak q_{target} of 10 MWm^{-2} and nuclear heating was performed. Results of the temperature and stress distributions showed that the design is acceptable with q_{target} ~10 MWm^{-2} under the detached divertor condition. Recent progress in the integrated conceptual design of the target units, coolant routing and cassette body will be presented.

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Iv-13: Predicting highly radiative divertor scenarios for fusion reactors

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Predictions for the scrape-off layer (SOL) in future fusion devices based on empirical scalings imply extremely large parallel heat fluxes, $q_{||}$ ~10 GW/m^2, making radiative dispersal of nearly all of the tokamak exhaust power mandatory to ensure compatibility with mature designs of solid, metal plasma facing components. Radiative divertor operation has been experimentally demonstrated on many devices, using a variety of radiating impurities (e.g., C, N_2, Ne, Ar, etc). Although radiative divertors have been widely demonstrated in existing devices, this solution must also be compatible with the (potentially conflicting) performance requirements for a future fusion reactor: for example maintaining high energy confinement and density lower than disruptive limits. Recent research has moved the focus from the scaling of the problem (e.g. $q_{||}$), instead looking at the scaling of radiative divertor solutions that rely on impurity seeding [1,2]. This combines scaling relations for the SOL width with those for other performance aspects (most notably the L-H transition power). This analysis shows that for H-modes, although the parallel heat flux increases strongly with toroidal field strength with little size scaling, the impurity fraction needed to reduce the divertor temperature to detachment levels increases more strongly with size than field, $f_Z \sim B_T^{0.88} R^{1.33}$. Further, while the needed plasma cooling can be achieved with finite ion dilution for a variety of next-step device parameters, this in general requires non-equilibrium ionization balance to increase the radiation for a given impurity fraction. New work has extended this analysis for an estimate using the L-I transition power and contrasted this with a scaling defined by the neutron wall loading.

While the analysis described above is useful for broadly studying how solutions to the heat exhaust challenge scale, it relies on a number of simplifications that should be improved on in order to develop reliable predictions. More sophisticated transport simulations using the SOLPS edge modeling code are being performed to improve on the analysis, in particular in the impurity transport aspects. Transport modeling allows physics-based calculations of the ionization balance of impurities, including in principle time-dependent effects, which are expected to play a major role in determining the final power dissipation. Similarly, impurity exhaust scenarios need to be compared with advanced divertors that feature strong neutral compression to achieve exhaust requirements [3] which avoid effects of impurity dilution. Kinetic neutral effects, including radiation transfer and well baffled divertor designs will be key to maintaining such scenarios. Further, the modeling provides a platform for more detailed benchmarking against experiment, for example to test the appropriateness of purely conductive heat transport as is often implicitly assumed.

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Iv-14: Strategy and results of high heat flux testing of European DEMO divertor mock-ups

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In the framework of the DEMO divertor project of EUROfusion an extensive R&D program has been conducted to develop advanced design concepts for the divertor targets. These plasma-facing components made of W blocks as plasma facing material and CuCrZr tubes as cooling tubes should allow a reliable DEMO operation for 2 h long pulses and maximum heat fluxes up to 20 MW/m². Compared to ITER, the operation at higher neutron fluence and the higher cooling water temperature of 150 °C is challenging and exceeds the currently existing experience.

Currently, eight types of unirradiated mock-ups manufactured during the 1st phase of the divertor R&D project are under high heat flux (HHF) examination in the test facility GLADIS at IPP Garching. The HHF test facility was equipped with a hot water cooling system to investigate the mock-ups under cooling water conditions similar to the expected DEMO operation. Each individual mock-up has to pass a screening test up to 20 MW/m² at 20°C cooling water inlet as first selection, followed by low cycle fatigue tests between 10 – 20 MW/m². Depending on the results, the thermo-mechanical investigation is continued using hot water cooling conditions for loading tests up to 300 cycles at 20 MW/m².

This contribution presents the strategy of HHF testing for the selection of the most promising concepts with limited HHF test resources. The influence of cooling water temperature, pressure and velocity on the heat transfer as well as the critical heat flux and the thermal behavior of the component are discussed on the basis of the HHF test results of the 22 tested mock-ups.

Iv-15: Tungsten Fibre-Reinforced Composites for Advanced Plasma Facing Components

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In view of the severe working conditions for plasma facing components (PFCs) in future power producing fusion devices the development of advanced materials is mandatory. The materials not only have to withstand high steady state power loads but also thermal cycling and shocks. Moreover, the change of thermo-mechanical properties by damage, activation and transmutation through fusion neutrons has to be taken into account when designing PFCs and selecting the adequate armour and structural materials. Within the research along the European Fusion Roadmap, water cooled PFCs are foreseen in a first DEMO design in order to provide enough margin for the cooling capacity and to only moderately extrapolate the technology developed and tested for ITER. In order to make best use of the water cooling concept copper (Cu) based alloys (as for example CuCrZr) are foreseen as heat sink whereas as armour tungsten (W) based materials will be used. Combining both materials in a high heat flux component bears the difficulty that their optimum operating temperatures do not overlap: W should be operated above 800° C in order to be in a ductile state to avoid brittle cracking under cyclic load, whereas CuCrZr should be operated below 300°C to provide enough mechanical strength. A remedy for both issues – brittleness of W and degrading strength of CuCrZr – could be the use of W fibres in W and Cu based composites. The W fibres used are drawn potassium doped tungsten wires as used in the lighting industry. They are characterized by very high strength (>2500 MPa), ductility already at room temperature and recrystallization and embrittlement only above 1900°C. Industrial textile techniques have been successfully established to prepare the W fibre preform for the production of flat tungs-ten fibre reinforced tungsten (W_f /W) samples and tungsten fibre reinforced copper (W_f /Cu) tubes. Since W and Cu do not form compounds or alloys and Cu easily wets W, the W_f /Cu composites can be readily produced by Cu infiltration into the W preform. Large samples of the W_f /W composites have been received by layered W chemical vapour deposition (CVD) and W chemical infiltration (CVI). In W_f /W composites the fibres must be coated by a temperature stable material (for example Y_2O_3) in order to avoid interdiffusion between W matrix and W fibres and to allow for the intended intrinsic toughening. The contribution will present the latest results on the production and characterization of the compounds, their measured and expected benefits and their integration into PFCs. In addition, it will point to important aspects concerning the plasma surface interaction with this kind of composites.

Iv-16: Powder metallurgically produced tungsten fiber-reinforced tungsten composites

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For the first wall of a fusion reactor unique challenges on materials in extreme environments require advanced mechanical and thermal properties. Tungsten (W) is the main candidate material for the first wall of a fusion reactor as it is resilient against erosion, has the highest melting point of any metal and shows rather benign behavior under neutron irradiation. However, the intrinsic brittleness of tungsten is a concern in respect with the fusion environment with high transient heat loads and neutron irradiation. Neutron induced effects e.g. transmutation add to embrittlement and are crucial to material performance. To overcome this drawback, tungsten fiber reinforced tungsten (W_f/W) composites are being developed relying on an extrinsic toughing principle. Accordingly, even in the brittle regime this material allows for a certain tolerance towards cracking and damage in general in comparison to conventional tungsten.

Tungsten (and other refractory metals) is industrially produced by powder metallurgical (PM) routes. Therefore, the development of PM routes is also important for tungsten composites in view of large scale production. Recent developments in the area of powder-metallurgical W_f/W will be presented showing a possible manufacturing path towards a component based on standard tungsten production technologies, such as field assisted sintering technology (FAST) and hot isostatic pressing (HIPing). W_f/W with ~94% and ~98% relative density can be produced by FAST and HIPing, respectively. The optimization of the chosen weak interface (Y_2O_3) between the fiber and matrix is crucial to realize pseudo ductility mechanisms.

Based on the initial mechanical tests and microstructural analyses, the PM-W_f/W materials can facilitate a promising pseudo-ductile behavior at room temperature. This enables the application of W_f/W in a fusion reactor where the temperature window between (at least) the coolant temperature and that of the plasma-facing surface needs to be covered. Even with brittle fiber and matrix, the as-fabricated samples show step-wise cracking while the material is still able to bear rising loads, similar to fiber reinforced ceramic composites. The crack propagation resistance is, therefore, significantly increased compared to conventional tungsten material. As plasma-facing material, the advanced crack resilience of W_f/W relies on the extra energy dissipation mechanisms, like fiber-matrix interface de-bonding, crack bridging by the fiber and fiber pull out, etc.

Iv-17: Outer divertor target heat fluxes during resonant magnetic perturbation induced ELM suppressed regimes in KSTAR

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One of the most important concerns for burning-plasma devices such as ITER is the power load on the divertor and its power handling capability. In ITER, the heat flux onto the divertor tile during the ELM crash would be around few 100 MW/m^2. This can impose severe limit on the lifetime of the divertor target tile. It has been demonstrated in several tokamaks that the transient heat flux due to ELM crash can be avoided by suppressing or mitigating ELMs with the external resonant magnetic perturbations (RMPs). RMPs are also expected to reduce the peak heat flux of steady-state divertor heat load by broadening the heat flux profile. It has been found in the L-mode plasma that the non-axisymmetric heat load can be spread out to all target tiles while the averaged heat flux in the toroidal direction is hardly changed by rotating the toroidal phase of RMPs. Thus, it is known that RMPs can act as an active control knob to exhaust the power onto the divertor target.

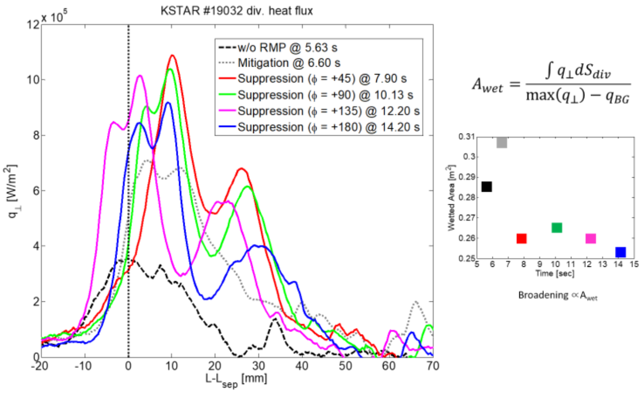

Fig. 1 Comparison of divertor heat flux profiles between non-RMPs, RMP-ELM mitigation and suppression regimes

However, it was firstly observed in the KSTAR tokamak that the divertor heat load profile becomes more sharpened accompanying larger peak heat flux in the ELM-suppression regime than those in the non-RMP and ELM-mitigation regimes regardless of the toroidal phase of RMPs as shown in fig. 1. This result may questions the application of RMPs as the power exhausting control knob since if the peak heat load during the ELM-suppression regime significantly increases above the engineering limit, we cannot take any advantage of the ELM suppression by RMPs. It also strongly implies that it is very important to understand why the heat flux profile significantly changes during ELM-suppression regime and find optimal RMPs configurations that can control divertor heat load effectively while suppressing ELMs.

Iv-18: High-confinement steady-state operation with quasi-snowflake magnetic configurations in EAST tokamak

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Advanced divertor configurations are considered as the alternative solution to power exhaust issues on tokamak fusion devices because of enhanced scrape-off layer (SOL) transport and an increased plasma wetted area on divertor target, especially during the long-pulse operation. Exact snowflake divertor (SFD) for EAST is only possible at very low plasma current due to poloidal coil system limitation. However, we found an alternative way to operate EAST in a so-called quasi-snowflake (QSF) magnetic configuration characterized by two first-order nulls with primary null in the separatrix and secondary null outside the vacuum vessel. Effective heat reduction was observed on EAST QSF diverted plasma with an increase of the connection length by ~30% and flux expansion in the outer strike point region by a factor of ~3, and this QSF configuration is suitable to the tokamaks with integrated poloidal field (PF) system like EAST, which wasn't original designed for the SFD configuration. A power flux reduction, up to a factor of 2-3, was achieved with bottom QSF on the graphite divertor. A mix of different auxiliary heating power, i.e., electron-cyclotron resonance heating (ECRH), low-hybrid wave (LHW) and ion-cyclotron resonance frequency (ICRF), has been injected up to total 6.2MW in upper QSF configuration with top tungsten divertor, without coupling problems. All the discharges were in H-mode with $H_{98} \geq 1$, neither edge D_{α} nor radiation burst or core impurity accumulation was observed. Fully non-inductive steady-state discharges have been achieved with all the main plasma parameters being very stable, in which the longest steady-state shots have been demonstrated up to 21s without any sign of instabilities [1], only limited by the technical imposed scenarios parameters. In all the steady-state QSF discharges, the ELM activity was quiescent, indicating a possible non-linear interaction between the downstream magnetic topology and the upstream kinetic gradients. High plasma density discharges (up to around 0.8 with respect to the Greenwald density $n_G$limit) have been achieved with a slightly lower confinement, probably due to the lack of additional power. This kind of high confinement steady-state ELM-free scenario offers one of the promising ways for future fusion reactors.

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Iv-19: Edge radiation control in stochastic magnetic field and with RMP application in LHD

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In LHD, edge radiation distribution and intensity control experiments have been conducted by changing thickness of the edge stochastic layer and by applying RMP field. Following experimental and modeling results were obtained.

1. The RMP application enhances edge impurity radiation and induces stable detachment. This is considered due to increase in volume of radiating layer caused by edge magnetic island. The VUV spectroscopy measurement indicates that the enhancement occurs at both X- and O-point of the edge island. The edge visible spectroscopy measurement shows that sustainment of enhanced low charge state emission of carbon (C^{1+}) and enhanced volume recombination, are kept outside of the confinement region. Divertor particle and power load estimated by Langmuir probe is reduced significantly with increased cross-field transport towards private region, and with toroidal modulation in accordance with the mode number of the RMP. Without the RMP, these radiation layer penetrates into confinement region and leads to termination of discharge. Effects of the edge magnetic topology (O- and X-point of island) on radiation stabilization is conducted by using simple model.

2. EMC3-EIRENE modeling of the RMP application case shows that selective cooling occurs at the X-point first, then moves to the O-point with increased density. These results are qualitatively consistent with AXUV and imaging bolometer measurements. But quantitatively, the simulation predicts higher radiated power intensity than the measurements by a factor of ~2 at the moment. It is also indicated from the comparison that the impurity transport (cross-field) coefficient is higher than that of background plasma at least by a factor of 4.

Clear change of density dependence of edge impurity emission intensity and distribution is observed for different thickness of the stochastic layer. It is found that edge Te flattening induced by the stochastic layer leads to stable partial detachment, when the flattening temperature enters ionization potential of C^{2+} and C^{3+} by enhancing the emission. The magnetic configuration without such Te flattening leads to sudden increase of radiation and leads to radiation collapse.

Iv-20: Heat and Particle Transport from the Stochastic SOL to Divertor in the Large Helical Device

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In this contribution, current understandings and unclear points of heat and particle transport from the intrinsic stochastic SOL to divertor in the Large Helical Device (LHD) are presented and discussed. Experiment data and the results of numerical analysis with EMC3-EIRENE are shown.

Understanding of the heat and particle transport in the stochastic SOL, and the transport to divertor is necessary to predict the heat and particle loads on the divetor tiles in the helical device and tokamaks with RMP. In LHD, the profiles of connection length of the magnetic field lines on the divertor tiles have more than one peaks like "lobe structure" in the tokamaks divertor with RMP. Therefore the understandings obtained in LHD can be applied to the prediction of the heat and particle load on the divertor in the tokamaks with RMP.

In the LHD divertor, multi-peaks of the heat and particle flux are observed. The position of the peaks are at the peaks of connection length profile. The ratio of the peak heights are modified by discharge conditions, such as, upstream plasma temperature. The reproduction of the profiles of heat and particle flux has been performed by using EMC3-EIRENE code, and in many cases, the profiles in experiment can be reproduced by the simulation. However some profile observed in experiment cannot be reproduced by the simulation. The effects of the variation of plasma temperature and the magnetic field strength in the SOL on the profiles of heat and particle flux were examined in the simulation, but the results suggest the effects are not so strong.

The asymmetry of the heat and particle loads on the divertor tiles, which are at the positions of symmetric magnetic field lines structure have been observed in the LHD experiments. The asymmetry inverts with the toroidal magnetic field reversal, suggesting that the asymmetry is caused by drift. At this stage, the drift effects cannot be treated with EMC3-EIRENE code.

Iv-21: Particle and Power Exhaust for H-mode Operation over 100 Seconds with ITER-like Tungsten Divertor in EAST

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A world record long pulse H-mode operation of 101.2 seconds with H_98=1.1 and a total power injection of 0.3GJ has been successfully achieved in the EAST tokamak with ITER-like top tungsten divertor, which has steady-state power exhaust capability of 10 MWm^{-2}. The peak temperature of W target saturated at t = 12 s to the value T≈500 ^oC and a heat flux ≈3 MWm^{-2} was maintained stably. Great efforts to reduce heat flux and accommodate particle/impurity exhaust simultaneously have been made towards long pulse of 10^2 s time scale. By exploiting the observation of Pfirsch–Schlüter flow direction in the SOL, the Bt direction with Bx∇B away from the W divertor (more particles favor outer target in USN) was adopted along with optimizing the strike point location near the pumping slot, to facilitate particle and impurity exhaust with the top cryo-pump. By tailoring the 3D divertor footprint through edge magnetic topology change, the heat load was dispersed widely and thus peak heat flux and W sputtering was controlled consequently. A number of effective approaches to remove core high-Z impurities, especially W, have been developed. By the exploration of high-frequency small-ELM H-mode regime, the impurities in the main plasma are effectively exhausted. On the other hand, with respect to the giant ELMs, the transient heat load and divertor W sputtering caused by small ELMs are well controlled. Similar results were observed with extensive lithium wall conditioning, in addition to lower edge recycling and low-Z impurity content. ECRH utilization, which is also very promising in expelling the high-Z impurities by tailoring the input power, will be also presented.

* See appendix of B. N. Wan et al., Nuclear Fusion **57**, 102019 (2017)

P-1: Snowflake diverted configuration design with in-vessel coils for CFETR

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Handling fusion power and particle exhaust, reducing heat loads below a limit on plasma-facing components, especially on divertor plates, are one of the critical issues for the long-pulse or steady-state operation CFETR and future fusion reactor. Advanced magnetic divertor configurations, like super-X divertor (SXD) and snowflake diverotr (SFD), are regarded as the alternative solution to power exhaust issues on tokamak fusion devices because of enhanced scrape-off layer (SOL) transport and an increased plasma wetted area on divertor target. The basic concept of the SXD has a divertor leg longer than the conventional diverotr, and it extends outboard in low field side, the divertor geometry and divertor coils arrangement needs more space which is not appropriate for CFETR. The SFD configuration, characterized by a second-order null or two first-order nulls (SF+ & SF-), with the presence of a large weak poloidal magnetic field zone by modifying the magnetic topology around the X-point, doesn't have the space concern. It will be a good choice for CFETR advanced shape experiments.

Equilibrium calculation code, F2EQ, have been developed for advanced divertor. The SF+ configuration was investigated in the first concept design phase of CFETR (R=5.7m, a=1.6m, B_T=4.5T, I_P = 8~10MA) [1], with 2 additional snowflake divertor coils (DCs) plus 12 integrated poloidal field (PF) coils locating outside toroidal field coil for the voltage seconds and plasma shaping. In the CFETR phase II (R=6.7m, a=1.8m, B_T=5~6.5T, I_P = 10~14MA) design, two in-vessel divertor coils were introduced for SFD shape design. Multiple SF+ equilibrium designs were carried out with different coil arrangement for I_P=10MA: one coil arrangement is 14 PF coils and 2 external DC coils (DC scheme), the other is 14 PF coils and 2 in-vessel coils (IC scheme). In DC scheme, the maximum coil current on 2 DCs is about 30kA/turn and the coil size is 1319.6x1319.6mm^2 with 324 turns. But in the IC scheme, the maximum coil current is about 12kA/turn and coil size is 370x370 mm^2 with 25 turns. In compared to the DC scheme, the flux expansion at X-point has a factor of 4 increase in IC scheme. Besides, there is more space in the low vertical portal which was proposed for the device assembling and maintaining in IC scheme. But there are some problems for IC scheme, like how to extend the lifetime of the in-vessel coils with so strong neutron irradiation. In the next step, systematical investigation of CFETR SFD configuration design will be carried out to obtain the optimized plasma shape, flux expansion, connection length, coil size and position and coil current etc.

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P-2: Conceptual Design of Magnetic Island Divertor in the J-TEXT tokamak

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Magnetic island divertor, which is characterized by the coexistence of closed magnetic surfaces, islands and open stochastic regions, has specific physics aspects different from the axisymmetric divertor. In the last decade, the physics of island divertor has been developed on stellarator devices, e.g. W7-AS [1], W7-X [2] and LHD [3], and on tokamaks like TEXTOR-DED [4]. Previous experimental and modeling results demonstrate that the 3D edge island geometry can cause momentum loss, and further effect on the edge particle transport and the edge recycling of divertor plasma. In addition, the connection length of the SOL field lines in a magnetic island divertor can be much larger than that in the poloidal divertor, and is expected to increase the thermal power decay length (λ_q) on the divertor targets.

On the limiter tokamak J-TEXT, the conceptual design of an island divertor configuration has been carried out. A set of saddle coils outside the vacuum vessel will be used to induce $n=1$ resonant magnetic perturbations at the plasma edge. The dominated magnetic component will be resonant at the q=4 rational surface, and an m/n=4/1 island divertor configuration will be used. In the design, the low m components, i.e. m/n=1/1 and 2/1, are optimized to reduce the effects on the core plasma and to avoid stimulating core islands or even causing locked mode disruptions. Both, the magnetic topology and the field line connection lengths have been calculated under vacuum assumption. The preliminary results show an m/n=4/1 island chain with an island width larger than 4 cm are expected to be obtained. A simplified divertor target for handling the heat exhaust has been also designed. In this paper, a detailed analysis of magnetic perturbation spectrum and some preliminary plasma transport calculations with the edge island divertor on J-TEXT will be discussed.

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P-4: Simulation Study of Quasi-snowflake divertor with Ne seeding

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To find an effective way to exhaust the heat power for future fusion reactor, snowflake divertor (SFD) [1] is thought as a possible candidate. In the snowflake divertor configuration, the flux expansion is significantly increased in comparison with traditional single-null divertor, which is of benefit to increase the wetted area and therefore decrease the peak heat load onto the divertor target. Meanwhile, the connection length and radiation volume are also increased in SFD configuration, which is considered of benefit to radiate the power in scrape-off layer (SOL). In our previous SOLPS simulation work [2], a reduction in the peak heat flux is shown for the quasi-snowflake (QSF) divertor in China Fusion Engineering Test Reactor (CFETR), in comparison with the lower-single-null (LSN) divertor.

In this work, neon is injected to achieve the radiative SFD. In the SOLPS simulation, neon impurity is seeded from the outer baffle and the puffing rate is scanned to have a ratio of radiated power to the power into the scrape-off layer (which is 100 MW in the simulation) up to ~ 80%. Along with the increasing of radiation power ratio, the effective charge number is increasing up to about 2.6-2.8. Furthermore, the outer divertor of SFD is found to have an abrupt transition to complete detachment, when the total radiation power is about ~ 50 MW. After the transition, the flow reversal in SOL of the outer divertor is almost vanished. Similar result is found for the inner divertor when neon is seeded from the inner baffle.

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P-5: Influence of the plasma parameters on the divertor power load for DEMO with tin divertor

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Aim of this paper is to investigate a divertor power load in DEMO reactor with liquid tin divertor dependence on perpendicular scrape-off layer (SOL) transport coefficient and density at the separatrix. The simulation is performed with the COREDIV code, which self-consistently solves radial 1D energy and particle transport equations of plasma and impurities in the core region and 2D multifluid transport equations in the SOL. Influence of the sputtering, evaporation and prompt redeposition is taken into account. The evaporation rate depends on the energy flux to the divertor target and the amount of tin released is, thereby, coupled with the plasma in the divertor region. The heat deposition due to re-condensation of Sn is presently neglected as the Sn vapour distributes the heat over a large area or is absorbed in the pumps. Increased heat flux leads to more evaporation and hence more Sn in the SOL. Vapors radiate and redistribute significant part of the energy flux to the plate creating a vapor shield.

First studies [3] show that DEMO operation in presence of tin impurity is not exluded although power across the separatrix (P_{SOL}) is slightly below the LH threshold, but still within the uncertainty limit. In the simulation described in [1] the D=0.5 m2/s value was used fort he radial transport in the SOL. Present studies indicate that higher radial diffusion (D=0.75 m^2/s) enlarges the operation space and P_{SOL} increases up to 140MW but at the expense of higher divertor power load.

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P-6: Active suppression of tungsten impurity influx using lithium aerosol injection in EAST

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EAST has upgraded the upper graphite divertor to ITER-like W/Cu monoblock structure [1] with active water cooling in order to facilitate the high power and long pulse plasmas [2]. Without wall conditioning tungsten impurity accumulation has been usually observed in plasmas, which is a crucial impediment to achieving high power, long-pulse H-modes. Therefore, some wall conditioning technologies need be explored to suppress the tungsten impurity, such as lithium (Li) aerosol injection [3] and Li coating [4]. In 2016, plasma discharges are performed in tungsten (W) upper divertor, and some exciting results are obtained with Li aerosol injection.

The real-time injection of a lithium aerosol into the edge of EAST discharges has been shown to be an effective technique to suppress the impurity influx from an ITER-like tungsten divertor. Moreover, aerosol injection has been systemically investigated in various operating scenarios, such as in both L and H-mode discharges and both before and after the deposition of relatively thick lithium evaporative coatings. During aerosol injection, several effects play a role in the suppression of tungsten influx. These effects include shielding by lithium ions at the plasma edge leading to a reduced scrape-off-layer temperature and a reduction of the heat flux on the tungsten divertor, mitigation of ELM size in the H-mode, and impurity segregation via deposition of a lithium film on plasma-facing surfaces. This research indicates that real-time lithium aerosol injection could be an effective method to suppress tungsten impurity influx, especially in steady-state operation of future fusion devices such as ITER or DEMO.

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P-7: Using Li to mitigate plasma-material interactions in EAST

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Mitigation of plasma-material interactions could effectively reduce heat flux onto the divertor plate and decrease, and reduce lots of impurity and fueling particles release from the wall. Li served as a low Z plasma facing material, has been shown to mitigate plasma-material interactions [1].

In EAST device, Li coating by oven, Li particle injections, and flowing liquid Li limiter (FLiLi) have been successively applied [2]. A new attracting physical phenomenon was observed that a bright Li radiative mantle at the plasma edge appeared during the Li powder injection and FLiLi operation [3]. Li powder was real-time actively injected into plasmas from the top of EAST device with a rate of 5×10^{21}atom/s（~70mg/s）, and Li particle passive efflux from Li limiter surface into plasma was estimated at $>5\times10^{20}$ atom s^{-1}, due to surface evaporation and sputtering, and accompanied with a few small Li droplets ~1 mm diameter that were ejected from FLiLi due to the strong interaction between the liquid Li surface and plasma. These Li particles would be ionized by SOL plasma and transport driven by SOL plasma flow to gradually become wide to form a bright Li radiative mantle with an obvious non-uniform space distribution. Significantly, strong Li radiation effectively reduced heat flux impacting onto the divertor plate to facilitate divertor heat flux removal and decrease the PFCs damage, with certain similarities to heat flux reduction and detachment onset via low-Z impurity injection. Moreover, Li injections and FLiLi experiments also have been successfully evidenced to mitigate transient heat fluxes impacting onto wall material surface by ELMs mitigation in H mode plasmas in EAST.

In future fusion device, by applying various Li injection methods, the amount of Li passive injection into plasma from liquid Li component installed at high heat flux zone, could automatically self-regulate Li efflux into plasma according to plasma impacting condition. By combining with Li active injection, it will be to effectively mitigate steady state and transient high heat fluxes respectively by Li strong edge radiation and ELMs mitigation to finally mitigate plasma-material interactions.

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P-8: Halo current, disruption mitigation on EAST with tungsten divertor

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Halo current induced in the scrape-off layer can flow through the first wall, which generates electromagnetic loads on the vessel and divertors, and is a great concern for tokamak. Therefore, accurate measurement and force evaluation are the key issues for the safe operation and upgrade of divertor.

The paper mainly describe: 1) the location of halo current diagnostics, 2) difference of halo currents flow between lower diver and tungsten divertor, 3) disruption mitigation with MGI and plasma soft-landing driven by lower hybrid current. The structure of halo current flowing on the wall mainly depends on the divertor structure, the EAST upper divertor has been upgraded with new tungsten divertor (monoblock and cassette) in place of the old “W” shape divertor. On “W” shape divetor DOME has born a heavy halo currents. However, on the new upper divertor halo current mainly touch the outer target, through cassette, finally back to plasma from inner target, less currents flow through DOME and water-cooling tube instead. The maximum halo current on the single divertor is up to 10 kA for 500 kA plasma current operation with TPF up to 2. The toroidal asymmetric current comes from the twist of plasma, it can be seen from saddle coil installed behind the passive plate, and the saddle coil can directly imply the distance between the plasma and the wall. In recent EAST campaigns, helium and argon have been used on halo current mitigation and get a nice mitigation results. Also we have found that lower hybrid current can continue drive disruptive plasma and lead to a soft-landing, finally result in less halo current, which is expected to be another effective way for mitigation instead of rapidly shutdown using MGI or pellet injection.

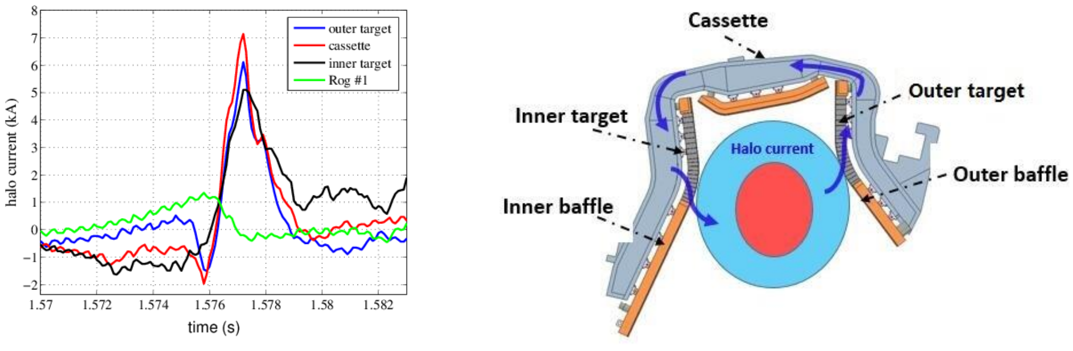

Fig. 1. The waveform of poloidal halo current on the single divertor module (left); Sketch of halo current flowing on the divertor (right)

P-9: Multi-Input Multi-Output control for EAST Quasi-Snowflake configuration

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Spreading the heat load at stick point one of the most impotant works for increase the plasma performence. The Snowflake divertor(SFD) configuration could increase the flux expansion near the stick points, which fact for spreading the divertor heat load [1] .Snowflake divertor configuration requires not only the first-order null at X-point as usual divertor configuration, but also the second-order derivative of the poloidal flux at X-point to be zero. Due to the Poloidal Field (PF) coils current limit, standard SFD configuration is difficult for EAST [2]. One Quasi-SFD configuration can keeps the current within the allowable range of the coils [3].

The shape control is an important part to keep the plasma configuration. In EAST, the shape control scheme is using 12 PF coils to control the flux of sevrial control points and the magnetic field of X-point. Due to the PF coils position, it is difficult to control each shape control points and X-point magnetic field by Single-Input Single-Output method. A numerical decoupling method for the Multi-Input Multi-Output (MIMO) system is required to solve this problem. The MIMO controller is designed with a physical based response model and the SVD method has been used to decouple the system. The preliminary results of MIMO plasma shape control for QSF configuration prove the potential for MIMO controller.

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P-10: Self-consistent kinetic trajectory simulation model of magnetized plasma sheaths

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Understanding of plasma properties in the immediate vicinity of a material wall is of crucial importance in all plasma applications. The plasma sheath plays an important role in determining the overall plasma properties as well as the particles and energies reaching the wall. We have developed a kinetic trajectory simulation model for bounded plasmas which is being used to study various situations of interest and yield results of high accuracy. The plasma-wall transition region in presence of oblique magnetic fields has been studied using the model. In presence of an oblique magnetic field, the electric field does not show the usual monotonic nature. Temporal dependence of ion velocities as well as their damping nature has also been studied. As we are using a kinetic model we expect that our model can help in understanding and resolving the physics of the scrape-off-layer region.

P-11: Overview of lower single-null neutral beam heated ELMy H-mode plasmas in TCV

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Recent experimental studies on the medium sized TCV tokamak, a carbon device featuring an open divertor, focus on the physics of the type-I H-mode in lower single-null plasmas. This regime, known to provide a good confinement but intolerable transient target heat loads in ITER, is now routinely achieved in TCV with the installation of a 1MW neutral beam injector. While pedestal parameters are estimated from the upgraded Thomson scattering diagnostics, heat loads at the inner and outer targets are measured by infrared thermography and complemented by Langmuir probe measurements. The inter-ELM heat flux decay length λ_q is of the order of 3 mm, about half the L-mode value, and it is in line with existing scalings, in particular $\lambda_q \propto B_{pol}^{-1}$. Surprisingly and in contrast with results from other devices, the estimated target fluency during ELMs is found to be a factor 5-20 lower than the predictions from empirical scalings. This might be a consequence of a larger SOL radiation due to medium Z impurities released from the neutral beam injector. In addition, impurity accumulation can trigger a 2/1 mode which strongly influences the estimated pedestal temperature and the heat load balance between the inner and the outer targets. A comparison with type-I H-mode from ECRH plasmas is foreseen to clarify our current understanding. It has also been demonstrated that a higher pedestal can be achieved by increasing the upper plasma triangularity δ_{up}, resulting in a better confinement. This improved confinement is also accompanied by a change in the ELM regime: small ELMs are observed in between the usual type-I ELMs whose frequency has decreased. A further increase of the density (f_G~0.7) is able to fully suppress the type-I ELMs but the impact on the global confinement is still under investigations. The effect of nitrogen seeding has also been investigated. It is seen that the pedestal pressure increases with the level of N2 injected but only for high δ_{up} plasmas.

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[†] See the author list in H. Meyer *et al* 2017 Nucl. Fusion **57** 102014

P-12: Understanding the role of pumping in the closed divertor

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Achievement of highly dissipative or detached divertor conditions to maintain both the deposited heat flux density below 10 MW/m^2 and plasma temperature at target below 5 eV is required for the future high-power steady-state fusion devices. Modeling results show that a closed divertor can increase neutral pressure and enhance radiative dissipation, thus it is proposed for advanced tokamak operation in order to achieve detachment at as low an upstream plasma density as possible. The recently experiment in Small-Angle-Slot (SAS) divertor developed by DIII-D also demonstrated the promising features of the closed divertor. However, the necessity to pump the closed divertor may result in reduction of the high density of neutrals, thus weaken the power dissipative capability of a closed divertor. In this work, the effect of pumping in a closed detached divertor configuration is examined with SOLPS modeling, which is a coupled version of the multi-fluid transport code B2.5 and the kinetic neutral transport code EIRENE. By changing the recycling rate at the pump of different locations, it is confirmed that the pump location has a significant impact on the effective pumping speed, which influences the divertor plasma significantly. The pump near the strike point region has the highest pumping efficiency. Higher pumping speed can reduce the neutral density and increases T_e as well as the heat flux to the target. For a given particle removal rate, however, the SOLPS simulation shows the plasma conditions are insensitive to the pump location within the divertor. By injecting heavy D_2 gas, detachment can be achieved and maintained even for the high pumping speed condition. In contrast, a deeper detachment can be easily reached in a low pumping and low puffing case.

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P-13: Pumping Optimization for the EAST Divertor Upgrade

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An upgrade to the lower divertor is currently being planned for the Experimental Advanced Superconducting Tokamak (EAST). The upgrade includes considerations of particle control through improved pumping geometry. Here we present an optimization of particle pumping for the divertor upgrade. The optimization is based on a semi-analytic pumping model initially developed and validated against DIII-D experiments [1,2], and later extended to better account for gas transport through ducts of finite length into the pumping volume [3]. This model, consisting of a first-flight neutral transport model combined with measured data for the divertor density, temperature, and particle flux, is valid for attached divertor plasmas where volume recombination is small. The model allows for a rapid analysis of prospective pump entrance locations and yields the optimal entrance height and length to give the strongest neutral pressure buildup.

For the optimization considered here, measured divertor profiles from previous EAST experiments have been used, mapped onto magnetic equilibria with separatrix strike point locations within the planned divertor. Pumping under a 'dome' in the private-flux region has been analyzed, with openings to either the inner or outer divertor leg or both. For the measured profiles used in the analysis, the particle flux to the inner divertor is much higher, and so only opening the pump to the inner leg gives the highest pressure in the pumping volume. On the outer divertor leg, a scrape-off-layer side pump entrance has been developed and optimized. In this case, adding a duct of length 5-10 cm allows pressures approaching 1 mTorr to be reached (several times higher than achievable in the case of PFR-side pumping). The sensitivity of the pump optimization to the width of the particle flux divertor footprint is being analyzed. Verification of the geometric optimization will be performed using SOLPS, which can be used to extend the analysis to the detached divertor case.

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P-14: Numerical investigation on the design space of a DEMO divertor: A quest for achieving high pumping efficiency

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One of the important design drivers of the DEMO divertor development is to ensure a desirable divertor performance for impurity removal. In order to achieve the aforementioned goal, a sensitivity analysis has been performed, aiming to improve the understanding how divertor performance is linked with the achieved pumping efficiency, defined as the ratio of pumped particle flux to the total sub-divertor incoming particle throughput. Since high density divertor plasma conditions are expected during DEMO operation, a reliable estimate of the macroscopic parameters requires a numerical tool to describe highly collisional flows. For all the above studies we utilize a new and efficient numerical tool called DIVGAS, which is based on the Direct Simulation Monte Carlo (DSMC) method [1]. The DIVGAS code has been used to model successfully the neutral gas flow in the JET [2] as well as in the JT60-SA sub-divertors [3].

The above analysis initially includes an assessment on the dome effect in a DEMO ITER-like divertor design within the framework of a free molecular approximation (collisionless case) [4] and under consideration of neutral-neutral collisions in the sub-divertor area [5]. It was found that the DEMO dome slightly contributes to the neutral compression in the sub-divertor and most essentially its existence adds a protection function of the x-point from the reflux of neutral particles from the private flux region. Additionally, the above sensitivity analysis includes an assessment on the divertor pumping performance for different pumping port sizes and locations [6,7]. It was found that the size and location of the pumping port may increase the pumped particle flux by a factor of two in the case of a large port positioned in the middle of the divertor cassette. Finally, an assessment of the size of inter-cassette gaps on the pumping efficiency of a generic 3D DEMO divertor configuration has been recently investigated [8]. The analysis shows that the pumping efficiency strongly depends on the inter-cassette gap width, and for the reference case of 20 mm gap width a reduction on the pumping efficiency up to 10% is obtained, compared to the case of a completely sealed divertor.

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P-15: Plasma simulation with SOLEDGE2D for complex divertor configurations on HL2M

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In a fusion reactor, heat and particle exhausts are primarily handled by the divertor system. There, surface components need to sustain very high thermal loads from plasma interaction, which has to be mitigated by volume radiation/dissipation. The need of extrinsic impurity injection to radiate is however also impacting the burning performances due to divertor leaking toward the confined plasma. Envisaged solutions to tackle this problem aim at optimizing the divertor geometry to increase divertor dissipation, improve impurity screening between divertor and confined plasma region, and improve detachment access and stabilization. Experiments showed that changing from limiter to simple X-point diverted geometry could significantly improve tokamak performances. Current developments investigate more complex geometries with multiple X- points in the divertor: snow-flake plus snow-flake minus, tripod, etc. An effective simulation code is required to properly estimate the power loads on the materials and understand diverted plasma specificities in terms of impurity screening and detachment access in these geometries.

SolEdge2D provide solutions for particle and energy transport in the edge plasma within complex and realistic 2D geometries. A penalization technique has been developed to model plasma-wall interaction in very flexible geometries. In addition, the plasma code has been coupled with the Monte-Carlo neutral code Eirene that implements the complex dynamic, atomic and molecular physics of neutrals within the plasma. This work makes SolEdge2D an efficient tool to investigate crucial problematics as main chamber recycling, impurity and divertor efficiency to control plasma wall interaction.

This work focuses on modelling of HL-2M scenarios, exploring different magnetic configurations: single null, tripod, snow flake plus and snow flake minus, constructed with EFIT code. The input power and particle fluxes at the boundary of the simulation domains have been scanned to find operating points in term of thermal load and target plasma temperature for the different configurations. Thermal loads on the plasma facing components, as well as plasma parameter profiles, are compared in order to quantify the benefit of each configuration: power spreading along target, dissipation, access to detachment. We examine the validity of using reduced models of heat transport in the scrape off layer to describe the results. Such models are indeed practical and commonly used guide the design and optimization of alternative divertor configurations.

P-16: Simulation Study of Divertor Detachment Density Threshold in EAST

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Divertor detachment is an effective and promising method to control the erosion and sputtering of divertor target materials for ITER and CFETR. It is necessary to achieve detachment at a relative lower upstream density to get a high current drive efficiency which is compatible with high duty cycle operations for ITER and CFETR. It is a significant work to find out what are the sensitive parameters and factors to affect the divertor detachment density threshold by modeling and experiment. In this work, we have studied the divertor detachment density threshold for L-mode discharge in EAST by using SOLPS code. The modeling results show that the suddenly increasing ratio of Dγ/Dα is in qualitative agreement with the experimental measurements on EAST when the particle flux landed on outer divertor target roll over. The effect of divertor closure on the threshold density has been investigated by modifying the location of the strike point at divertor target. The experimental and modeling results show that a good divertor closure is helpful for the capture of neutral particles to lower the density threshold of detachment. In additional, the predictive modeling works on the effect of plasma current (I_p) and the power across the scrape-off-layer (SOL) P_{SOL} on the detachment density threshold have been studied. We also find that the density threshold can be approximately scaled by $P_{SOL}^{5/7}$. Further work about the experimental validation of the predictive results will be carried out on EAST.

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P-17: Radiative Divertor for CFETR by Impurity Seeding

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China Fusion Engineering Test Reactor (CFETR) [1] is proposed to bridge the gaps between the fusion experimental reactor ITER and the demonstration reactor (DEMO). CFETR will be operated in two phases, for which the fusion power are ~200 MW and >1 GW, respectively. The new design parameters, *i.e.* R = 6.6 m, a = 1.8 m, B_T = 6-7 T, are chosen to meet both Phase I and Phase II targets and also to save the cost of construction. For Phase II, due to the high fusion power, heat exhaust would be a serious problem for long-pulse or steady-state operation.

To reduce the heat load on divertor, impurity radiation is effective and indispensable for CFETR. However, to reduce the tritium retention and increase the lifetime of plasma-facing components, full-W wall would be the prior choice, which means there is not any intrinsic radiative impurities. Therefore, impurity seeding is necessary to solve the heat exhaust problem by radiative divertor. Meanwhile, the seeded impurity has to be screened from the core plasma to avoid degradation of the performance.

In this work, to understand the tradeoff between two key parameters n_e and Z_{eff} for certain radiation power P_{rad}, SOLPS simulations are performed for various kinds of impurities, seeding locations and divertor geometries. The result is found in good agreement with Matthew law [2]. It could be considered as the boundary condition for core plasma simulation.

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P-18: Simulation Study of local closure effect of the Divertor structure on Detachment Density in CFETR

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The need for advanced divertor solutions to efficiently dissipate heat from fusion reactors is critical because the maximum steady-state power load is limited to $q_t \leq 5– 10\ MWm^{-2}$ to a plasma facing component (PFC) target surface, whether solid or liquid, while the undissipated power loads will be an order of magnitude higher. Previous work shows that the tightly closed divertor greatly improves trapping of recycling neutrals, thereby increasing radiative and charge exchange losses in the divertor and reducing the electron temperature T_{et} and deposited power density q_{dep} at the target plate. We studied the local closure effect of the divertor structure on the detachment density in CFETR by using the scrape-off layer plasma simulation (SOLPS) code. The divertor geometry near the striking point was set as a semicircle structure. The local closure of striking point was changed by reducing the radius of the semicircle step by step with all the other parameters fixed the same. The simulation results reveal that the heat flux show great differences in low density case and high density case. In low density case, the applied of semicircle structure (r~10cm) can hardly change the heat flux of the striking point. Yet, for the high density case, the semicircle structure will greatly reduce the temperature of the striking point, thus enhance the density of the neutral gas. In other words, the increased local closure effect will significantly reduce the heat flux of the striking point and help reduce the threshold density of the detachment.

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P-19: Evolution of impurities and radiation distribution in the edge to boundary region of tokamak plasma towards divertor detachment

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We look into 3 questions. How does the confinement of low- or medium-Z impurities inside the divertor change with different divertor plasma conditions? How does the impurities radiation distribution look like meanwhile? How may transport affect the impurities cooling efficiency?

The radiative divertor experiments with Ar seeding on ASDEX Upgrade showed that the divertor enrichment increased with the increasing throughput of Ar at otherwise similar operation parameters. Consistently, a series of modeling using 2D transport code to simulate the boundary plasma demonstrated the increase of divertor compression of Ar with increasing throughput of the impurity before the outer divertor got pronounced detachment. Modeling showed that the mean free path of neutral Ar was less than 1% of the divertor leg length in the outer divertor. It increased from less than 0.7% of the divertor leg in attached condition to 6% in detached cases in the inner divertor towards detachment. The net force of frictional and thermal forces on Ar species became stronger and pointed to the target in the inner divertor region towards detachment. Modeling reproduced the movement of the strongest radiation region along the divertor legs. Furthermore, modeling showed that radiation mainly increased outside the divertor with increasing throughput of Ar. We found that this attributed to the increased radiation efficiency of impurities outside the divertor. Inside the divertor, the cooling efficiency decreased. Comparing the cooling curve from atomic database and the one calculated from modeling results, we found that the latter diverted from the former, with the cooling curve decreases monotonically with Te at Te > 40 eV. This is an effect of transport processes.

P-20: Comparison of Ar and Ne radiative divertor plasmas in EAST

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A critical issue for ITER and future fusion devices is the handling of excessive heat load. It will require a high radiation level for power exhaust to avoid divertor heat overload and excessive surface erosion rates. Increasing divertor radiation by injecting impurities is a general and effective method to reduce scrape-off layer heat flux and to cool the divertor plasma to detachment. In recent years, argon (Ar) and neon (Ne) have been widely used in radiative divertor experiments on EAST and other devices. To compare effects of these two impurities and understand well the radiative divertor physics, both experiment and simulation work of Ar and Ne seeded plasma are carried out simultaneously in EAST.

Based on a typical H-mode partially detached plasma, we took relevant modeling by using SOLPS 5.0. The simulation results of plasma parameters agreed well with measurements. Both Ar and Ne can effectively reduce heat flux at divertor targets. But they show different radiation characteristics: Ar can reach higher radiative fraction than Ne, and it radiates strongly in divertor region and inside the separatrix, while Ne radiates mainly around X-point and separatrix. The experiment results also demonstrated the high efficiency in reducing heat flux and heat load of Ar and Ne. After these impurities seeded from divertor region, saturation ion current, I_s, electron temperature, T_e, and heat flux, q_t, on divertor target, decreased clearly. Completely and partially detached plasmas were also achieved. In this case, we obtained over 15s long and radiation fraction, f_{rad}, over 40% pulses with radiation feedback control under divertor gas puff and SMBI collaborative working. In addition, same as simulation predicts, gas puff methods and locations also affect the radiative divertor behavior. However, in contrast to the simulation results, there are significant differences in experimental radiation distribution. Radiation after Ar seeding mainly distributed in core region, but in Ne seeded situation, it is more likely to distribute in the divertor region. After Ne seeding pulse terminated, radiation remained in a rising state until plasma burned out, while the radiation dropped down gradually after Ar seeding. The reason could be that Ar has higher cooling factor than Ne near separatrix and separatrix inside. Moreover, compared with Ne, Ar is a high-Z impurity, so it is easier to cause physical sputtering on divertor targets. Sputtered impurities, such as Li, C and W, play very important roles for power exhaust in EAST discharge. These impurities presented a much higher level during Ar seeding than Ne seeding. Some of them, especially for tungsten, are transported into core region and led to the rise of core radiation. It is indicated that using Ne as the radiator preferentially in controlling PWI issue in radiative divertor experiments in EAST. But for the future devices, Ar should also be considered due to its high cooling efficiency. Therefore, considering the transport and enrichment process of impurities, more simulation tools will be used for next step works.

P-21: Impact of divertor configuration on H-mode access and confinement on EAST

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A good control of plasma wall interactions is of great importance to obtaining and maintaining high energy confinement in high-power long-pulse discharges. Future devices will require a high scrape-off layer (SOL) density to avoid high-Z sputtering. However, some parameters such as divertor configuration are not included in existing scaling laws, making extrapolations to future larger devices difficult. It is vital to understand the impact of divertor conditions on H-mode plasma in order to make engineering designs and operational scenarios most favorable to high fusion gain.

EAST upper divertor has been upgraded into ITER-like W/Cu mono-block structure in the target plates. Moreover, the new W divertor has a different geometry compared to the lower divertor, facilitating higher triangularity plasmas. Experimental comparison shows that the plasma with higher upper triangularity (δ_U =0.6) in upper single null configuration has a lower H mode power threshold and higher energy confinement compared to the counterpart plasma with a lower triangularity (δ_U =0.55). The change in divertor particle exhaust due to different outer strike point (OSP) positions should play an important role. The high triangularity plasma with OSP closer to the pump slot between outer target plate and dome has a lower impurity content both in edge and core plasma. Moreover, two plasmas also present different edge density profiles, which could be related to the different divertor D particle recycling. H-L back transition happens earlier in low triangularity plasma at the same density ramping rate. A significant outward shift of the edge density profile was observed during the back transition in both plasmas. Furthermore, statistic investigation shows that the L-H transition power threshold presents a dependence on the ratio of upper and lower triangularity, indicating the important role of main plasma shape, which needs further study.

P-22: Radiative feedback control for power exhaust in EAST long-pulse operations

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The excessive heat loads on divertor targets is one of the most significant challenges for tokamak development. In order to solve this issue, many tokamaks, such as ASDEX [1], have explored and implemented the active control of radiation. By seeding some minor impurity gas (neon, argon, etc.) into vacuum, a series of radiative processes exhaust heat power effectively [2], reduce the heat flux strikes to the divertors. Based on current EAST experimental condition, the active feedback control system of radiated power has been set up and employed in actual plasma discharges.

The radiative feedback control system on EAST is including three main parts. The supersonic molecular beam injection (SMBI) [3] located at the midplane and the piezoelectric (PE) valves located at divertor target plates are used to impurity injection as feedback and feedforward actuators respectively. The real-time radiated power is calculated by the signals from AXUV diagnostic [4], and the algorithm to do PID operation and send control command is in the EAST Plasma Control System (PCS) [5]. In the EAST long-pulse steady-state discharges with tungsten divertor in 2016, a reliable control capability is achieved. Either in L-mode or in H-mode, the total radiated power can be maintained on the pre-set target value. The control error is acceptable and no control instability is found for now. Meanwhile, even if the radiated power of core plasma is increased, the stored energy of main plasma has no significant decline, which means the plasma performance do not degrade seriously. The temperature of divertor target plates are in a low level during the radiative control phase, while the particle flux towards divertor target is also well controlled.

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P-23: EAST lower tungsten divertor concept and technics development

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The EAST upper divertor was updated to tungsten divertor in 2014, but the lower divertor was still left carbon divertor. The upper divertor heat load exhausting capability can be up to $10MW/m^2$, but only $2MW/m^2$ for the lower divertor. To meet the device high performance plasma operation and high plasma heating power. The lower divertor planed to be updated to tungsten divertor.

The lower tungsten divertor concept design started from 2016. Considering plasma configurations, neuter particles exhausting, field expansion, cost reduction and so on, geometry of lower divertor was optimized and will be different from upper tungsten divertor geometry.

ITER divertor monoblock structure is very good for high heat load exhaust, but manufacturing is quite complicate and cost quite high. New cooling structure and new manufacturing technologies are developing for the EAST lower tungsten divertor. Explode welding technology applied to combine material of CuCrZr and stainless steel. Cooling channels can be fabricated before exploding weld. Cross section of the cooling channel can be fabricated square or hypervapotron, and cooling efficiency increased much compare with circular cross section. Stainless steel mechanical properties do not affect very much by temperature. It is not very important to recovery CuCrZr mechanical properties after high temperature technics process. In this case if tungsten brazing to CuCrZr/SS unit stainless steel would act as mechanical support for the unit. Besides braze technology a kind of new diffusion welding technology was developed. It is different from HIP technology. The diffusion technology does not need very high pressure between tungsten and CuCrZr, and equipments for welding are not so complicate. Compare with monoblock structure new structure and new manufacturing technologies expect decrease divertor cost ~30%.

P-24: W coatings on divertor secondary areas

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The most advanced Plasma Facing Component concept for the future tokamaks (including ITER and DEMO) is the well-known W monoblock. This type of PFC is able to handle from 10MW/m² to 20MW/m². However, the divertor areas far from the strike points should have lower heat flux requirements, and may be designed with a different concept.

The ITER-like W monoblocks will be tested in realistic plasma environment in the next months in WEST (W Environment in Steady-state Tokamak), to mitigate the risks for ITER. In addition to the use of the ITER like divertor, the other PFCs in WEST have also been redesigned. In particular solutions with W coatings have been used to transform Tore Supra into a fully metallic environment.

The company DEPHIS has been chosen to carry the R&D on W coatings onto actively cooled PFC made of CuCrZr, using PVD (Physical Vapour Deposition) process. Specifications are defined in terms of density (>90%), homogeneity, impurity content (lower than a few percent) and absence of cracks and scratches.

After final validation of the parameters on the samples, a representative mock-up has been coated. It has then been qualified under thermal loads at GLADIS facility (IPP Garching) with 10MW/m² and 500°C. No defects appeared during the test, showing the robustness of the process considering the main source of thermal loads.

However, damages on coatings have already been observed in WEST during assembly due to manipulation. Moreover, erosion or even delamination are likely to appear sooner or later due to the harsh plasma environment, in particular with the energetic transient events.

A project newly started at IRFM focuses on the in-situ healing of these coatings. The foreseen process will be divided in three phases: ablation and cleaning of the damaged area using a laser, control of the cleaned area with Laser Induced Breakdown Spectroscopy (LIBS) and coating of the area using High Power Impulse Magnetron Sputtering (HiPIMS) process. These three devices will be miniaturized and mounted on an already existing robotic arm, able to reach any location inside WEST.

Several challenging issues have been solved during the R&D on coating W onto a CuCrZr substrate, in particular handling complex shapes of the PFCs. WEST operation will allow assessing the performance of the newly developed W coatings under tokamak environment (transient loads like ELMs, disruptions…). R&D now is focusing on the in-situ healing process of theses coatings using a robotic arm.

P-25: Conceptual design of the new water-cooled W/Cu lower divertor in the EAST superconducting tokamak

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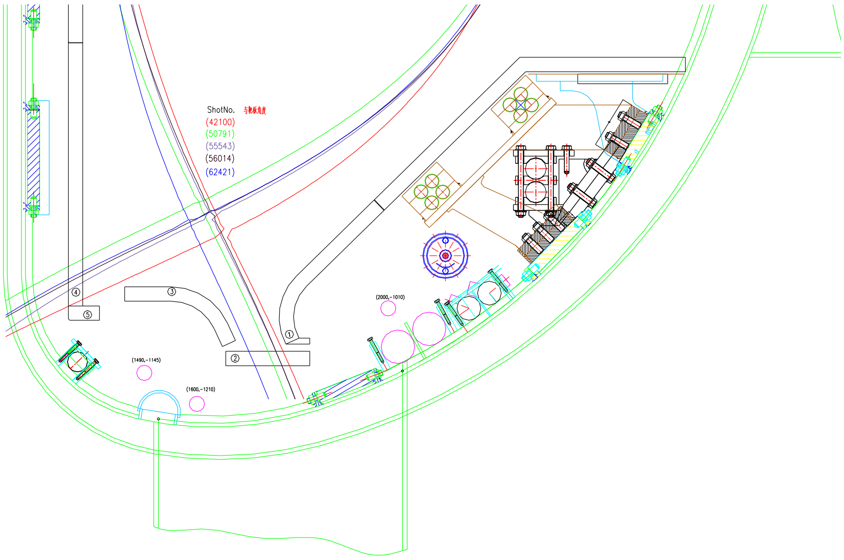

Fig.1 Conceptual design of the EAST new water-cooled W/Cu lower divertor

EAST superconducting tokamak aims at >400s high-performance H-mode operations with a full metal wall and a divertor steady-state heat load of ~10MW/m^2, demonstrating the physics basis of the Chinese Fusion Engineering Test Reactor (CFETR), the next-generation tokamak currently under R&D in China. Divertor heat exhaust and particle control is one of the keys for achieving long-pulse H-mode operations. However, the power handling capability of the lower graphite divertor is presently ~2MW/m^2. Hot spots currently limit the total heating power to <3MW for ~100s long-pulse operations. In addition, the pumping capability of the upper ITER-like W/Cu divertor is limited. Therefore, a new water-cooled W/Cu lower divertor is being designed (Fig.1). This new divertor employs a simpler geometry to facilitate manufacturing and engineering quality control, including a horizontal outer target plate and a vertical inner target plate that accommodates a wide triangularity range (0.4-0.6), which should allow access to small-ELM regimes. A more closed outer divertor geometry is employed and the pumping slots are optimized to maximize the neutral pressure and pumping speed, targeting at steady-state radiative divertor operations. SOLPS simulations indicate that this design can effectively reduce tungsten sputtering from the strike points as well as the far scrape-off layer. The monoblock structure will be used in the horizontal and vertical target plates, which has been demonstrated in the EAST upper divertor, allowing a power handling capability of ~10MW/m^2. The W-tile (2mm thickness)/Cu welding structure will be used in the low-heat-load area, e.g. the dome and baffle, allowing a power handling capability of ~5MW/m^2. The dome and baffle are oriented to avoid directly exposing the water-cooling end boxes to the plasma heat flows. The water-cooling system will be upgraded to accommodate 8m/s flow velocity and total 800t/h flow capacity. Three water-cooled internal coils are planned (Fig.1) with DC current up to 10kA for each coil, to achieve quasi-snowflake magnetic configurations in steady state with magnetic flux expansion increased by a factor of 4 at I_p = 0.35MA.

P-26: Theoretical analysis on the damages of tungsten plasma facing component under both steady-state heat loads and transient heat fluxes in fusion devices

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In ITER and future fusion devices, the plasma facing components (PFCs) are expected to suffer from the superposition of continuous steady state and pulsed transient thermal loads. The high transient heat flux in ITER could cause the material surface deterioration that increased from roughing to cracking [1 - 4]. In addition, the operating temperature of tungsten materials caused by the steady state thermal heat load could also have great influence to the cracking threshold [1 - 2], because of the high ductile-brittle transition temperature (DBTT) of tungsten. Therefore, it is important to assess the cracking failure mechanisms of tungsten under the superposition of continuous steady state and pulsed transient thermal loads.

In this paper, a finite element model based on a simplified ITER-like W-Cu module was built using direct coupling analysis and considering the temperature-dependent parameters of materials. The surface temperature and stress-strain distribution as well as evolution of tungsten under the superposition of steady state (0 - 3.7 $MW \cdot m^{-2}$) and transient heat load (5ms, 0 - 600 $MW \cdot m^{-2}$) were obtained by this model. According to the criterion which was the appearance of surface plastic deformation under DBTT during the cooling stage, the relation of the cracking threshold and the base temperature were acquired. It was found that when the base temperature exceeded the DBTT, the criterion of cracking cannot be satisfied which means the surface cracking of tungsten may not happen and the results. The high heat load test experimental results [3, 4] also have the same tendency. In addition, this phenomenon was explained by the analysis of stress and strain in this paper.

This work presented a clear thermal mechanical evolution process for tungsten under fusion relevant steady state and transient heat flux. The results of finite element analysis explained the relation between cracking threshold and base temperature on the respect of thermal stress and strain. According to the analysis, this work also gave a possible explanation and provided a feasible finite element calculating method to the cracking threshold in different base temperature.

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P-27: Failure analysis of EAST Lower divertor

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Experimental Advanced Superconducting Tokamak (EAST) has equipped the lower divertor since 2008. Lower divertor comprises total 2952 graphite tiles, 96 heat sinks and 48 support frames. With the length of plasma pulse in H-mode reached 100s thermal power deposited on the divertor has become very larger. Lots of hot spots had been observed on the lower divertor while plasma discharge. During plasma discharge tens graphite tiles damaged and a lot of dust of carbon made the plasma quench at last. By the analysis there are three main failure modes for graphite tiles: crack, erosion and release. A few graphite tiles cracked into two or three pieces and most graphite tiles had been erosion on the corners, edges and front surfaces. Some graphite tiles have no any damage on them but their position changed. Here the damage process has been researched by simulation and experiments to find the failure mechanism. Most failures are because of overheating. When the high heat flux on anywhere of graphite tiles expired the limits of lower divertor the erosion occurred some failures are because of electromagnetic force. When electromagnetic force on the graphite tiles was larger than the strength of graphite the crack occurred. Other failures are because of both thermal power and electromagnetic force. When the graphite tiles were heating the preload to mount the tiles reduced largely. The electromagnetic force would pull or push tiles to make the position of graphite tiles changed again and again and then they became released from heat sink. The failure analysis here helped to realize the performance limits of EAST lower divertor and helped to optimize or upgrade it in future.

P-28: Investigation on the Thermal Conductivity and Heat Flux Route of the Tungsten Divertor Monoblock under the Fracture Conditions for EAST Tokamak

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EAST tungsten divertor has experienced the plasma discharges with fruitful outcomes at the long-pulse H-mode for 4 years. The divertor failure issues on the tungsten monoblocks are concerned on recently for the high parameter operations in the plasma discharges. In this work, the fractures and the cracks issues on tungsten blocks are focused on since it has the great impact on the divertor thermal conductivity. Numerical simulations based on finite elemental method are carried out to confirm the change of the thermal conductivity properties. It is found that the heat transport route is interrupted due to the subtle structure change and the heat flux transmits along with the tungsten block side to the bottom area. This is crucial event which takes a very strong negative impact on the tungsten block because of the temperature rise rapidly in the bottom region. It indicates that the temperature rise on the surface facing to plasma side is lower than that of in the bottom region. This will lead to the tungsten surface facing to the plasma side will be accelerated to assume the damage risk on the surface material fracture in the heat fatigue cycles. In addition, much more thermal stress accumulation will be occurred in the area of high temperature region, which will affects the divertor structure safety including the supports. For further research, a detailed experimental scheme is proposed to explore the failure mechanism of the tungsten divertor for EAST tokamak in the future.

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P-29: Fatigue and fracture analysis on ITER-monoblock heat sink at operating temperature

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The most advanced mature plasma facing unit (PFU) technology is the ITER W/Cu PFU, which is a monoblock structure composed by tungsten, copper and CuCrZr as plasma facing material, interlayer and heat sink, respectively. Fatigue failure in heat sink structural including fracture under operating condition greatly influence the lifetime of divertor target. To investigate the fatigue behavior in CuCrZr tube in monoblock, the thermal-stress analysis was performed firstly with regards operating and cooling condition. Then, a quasi-static compression test on CuCrZr in divertor working temperature range was performed to obtain basic stress-strain relationship using MTS hydro-servo system. The strain- life relationship was deduced to assess the fatigue life of CuCrZr tube. As for the CuCrZr tube takes a risk of cracking and fracture under long pulse operation in future fusion device. The *finite element analysis and J*-integral *were applied to investigate the crack and fracture* performance *due to irregularity* joint surface such as roughness and defect in assembly and manufacture which would provide the necessary theoretical basis for fatigue evaluation of divertor target.

P-30: Status of technical R&D activity for EAST lower tungsten divertor

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Technical R&D is essential for optimized design of the EAST lower tungsten divertor planning to finish assembly in the mid-term of 2019. This poster presents the most urgent technical R&D work that had been recently finished or is expected to be launched in the near-future. In order to improve quality control of contact or welding bond between the armour and the heat-sink of the divertor prior to its installation into the vacuum vessel, a non-destructive examination method by transient infrared thermography is being developed and had been applied to test the lower graphite strike target. To likely use the Mo material in the lower heat flux area, the structure of Mo tile with bolt was tested by electron beam testing facility. The result shows the structure can sustain the heat flux of more than 2.5 MW/m^2. Later, a welding structure using resistance welding with solder method is being carried out to raise the capacity of heat flux and increase the interface strength. To eliminate the weld lines between CuCrZr plate and 316L tubes, a new explosive welding method manufacturing the hollow plate of a typical CuCrZr/316L heat-sink plate will replace other welding methods like HIP, laser or electron beam. The explosive welding structure manufactures basically through the process of milling the groove, filling the mould and the explosive welding, and analyzes the interface by mechanical shearing, bending test, bond interface analysis and SEM scanning, and then tests the pressure resistance of flow paths within. To find a more economic or a new method to fabricate the tungsten flat tile or monoblock structure, the explosive welding and direct bonding differ from the HIP method is being studied. The explosive welding of tungsten is an innovation method relies on brittle ductile transition theory of tungsten and suitable constraint of specimen in the explosive process. The direct bonding method without using interlayer metals is used to construct the metallurgical interface directly between Cu and W and make the tensile strength of Cu/W bond close to the Cu.

P-31: The Engineering Design of EAST Lower Advanced Divertor

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In the next 3-4 years, the objective of EAST Tokamak is to achieve longer pulse more than 400s H-mode plasma. So the divertor system of EAST shall accommodate the high heat flux up to $10MW/m^2$ on the divertor target. The upper divertor now is tungsten divertor with mono-blocks structure. But the lower divertor of EAST is still carbon divertor, which can only endure heat flux up to $2MW/m^2$. So the lower divertor shall be updated to fit such heat load. To get snowflake and others advanced divertor shapes, three divertor coils are configured under the lower divertor.

The whole lower divertor is divided into 48 modules. Each module is composed of an inner target plate, a Dome, an outer horizontal target plate and an outer vertical baffle plate. To minimize the total mass flow of the cooling system, all these plate are serially connected. To optimize the heat removal capacity of the cooling structure and cooling pipe system of the lower divertor, several types of cooling structures and cooling water distribution systems are considered. Through thermal analysis, fluid analysis and results comparison, the potential final cooling system is determined.

To support and bear the large electro-magnetic forces on the divertor module, the support systems of the lower divertor are designed. The support is composed of a cassette and an independent support leg, which is convenient for installment and maintenance. The structure of the cassette is a hollow-centered rectangular frame with wing plates on upper both sides, which can bear the loads from all the directions.

To justify the whole structure, the related thermal and structure analysis are carried out on the whole divertor module. The required flow rate of cooling water is applied in the cooling channels. And the related electro-magnetic forces are distributed on the structure. The results show that the whole structure can remove the heat flux up to $10MW/m^2$ and bear the large force due to the halo current during the plasma disruption.

For the manufacture processes, the relevant R&D work, such as vacuum brazing and diffusion welding of tungsten tiles and tungsten blocks, is performed on the first wall components and related thermal tests are done on the samples and prototypes, which will lay a solid foundation for fabrication of the lower divertor.

P-32: EMC3-EIRENE modelling of 3D impurity transport in argon seeded EAST discharges

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The transport property of impurities in the edge plasma is one of the most critical issues for the operation and performance of the fusion devices. The radiation of impurities during penetration into the confinement region of the plasma would lead to the degradation of energy confinement and even the termination of plasma discharge, which imposes constraints on routine operation of fusion facilities. Moreover, transport of eroded impurities leads to the build-up of tritium-rich layers in the shadowed and remote areas, which has a stringent limit on the viability of a future reactor due to safety and economy considerations. However, on the positive side, a good control of the edge impurity radiation would lead to a strong reduction of power load on divertor target plates. Therefore, studies of edge impurity transport are important to obtain a better understanding of the underlying mechanisms of impurity screening and detached divertor plasma.

The external argon (Ar) gas puffing experiments have been conducted on EAST tokamak to study the edge impurity transport and radiative divertor plasma. The transport behavior and line emissions of the Ar gas puffing in the scrape-off layer (SOL) of EAST have been simulated by the three-dimensional (3D) edge transport code EMC3-EIRENE [1-3]. The chord-integrated line emissions of Ar impurity are calculated by a sophisticated post-processing program [4-5], which can trace the lines of sight for each observation chord of the spectrometer through the 3D emission distribution obtained from the EMC3-EIRENE code. The impact of the gas puffing positions, which are located at the dome, in- and out-board divertor in EAST experiments, is studied to check the efficiency of the radiative power exhaust for the different poloidal gas puffing locations. Further, the Ar gas injected at three toroidally-localized puffing nozzles have been simulated to investigate the 3D effect of impurity transport in the SOL. In addition, the impact of the Ar impurity radiation on the reduction of the power load on divertor target has been analyzed.

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P-33: 3D modelling of tungsten fuzz growth under the bombardment of helium plasma

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Tungsten has been widely considered as a primary candidate material for the plasma facing components in next-step fusion reactors due to its high melting point, high thermal conductivity, low sputtering yield and tritium retention. However, experiments performed in linear helium-containing plasma devices indicate that a fiber form nanostructure, so-called "fuzz", is generated on tungsten surface under certain conditions: the surface temperature is from 1000 to 2000K and the incident ion energy is higher than 30 eV. The diameter of each tendril is roughly 10-50 nm and up to micron in length [1]. The nanorods structure dramatically change the tungsten morphology which leads to the decrease of optical reflectivity [2] and thermal conductivity by several orders of magnitudes [3]. It also can cause the enhancement of tritium retention and tungsten release. In addition, the unipolar arc on the tungsten fuzz surface is easily ignited in response to the pulse irradiation [4]. Thus, the mechanism of the growth of the tungsten fuzz and its properties are needed to be studied.

In this study, a three-dimensional (3D) kinetic Mont Carlo (KMC) code, SURO-FUZZ (upgrade version of SURO code [5-8]) has been developed to investigate the formation of fuzzy nanostructure in micro- and second-scales. The time evolution of fuzz morphology under Helium bombardment is studied by SURO-FUZZ. The growth rate of tungsten fuzz shows a good agreement with the experimental results.

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P-34: Numerical Study on Dust and Droplet Transport from Tokamak Divertor

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A plasma is a medium full of free charge particles. If dust and droplet, both of which are defined as a solid and a liquid macroparticle respectively, submerge in the plasma, those charge particles sink on them, so that they are charged. Dust- and droplet-plasma interactions are complicated but pose critical issues on the efficacy of fusion operation, e.g. health and safety issue on radioactive and chemical toxic carriers, radiation cooling, dilution of fuels etc.

To understand their behaviors, which can be further used for the anticipation of the fusion efficiency, we have selected the computational way for the study. The (first) simulation code of dust transport in a plasma, being developed in Thailand at present, consists of the physical models of charging and heating by plasma and macroparticle motion especially by electromagnetic force. Additionally, the detail of impurity release due to vaporizations controlled by both electric and pressure are considered. The main aim is to simulate a test W and Be dust or droplet, which are presumably initially released from divertor region, from which W dust and droplet are newly generated, e.g. during inter-ELMs, and in which W and Be dust of former operations were accumulated. This is in order to understand dust or droplet parameters related to, e.g. velocity, life-time, end-up position, impurity generation, radiation etc., at separatrix as well as in SOL. Subsequently, those parameters should help us better understanding the impurity transport and their effects on a fusion plasma in both edge and core regions.

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