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Programme & Book of Abstracts

9th IAEA Technical Meeting on Steady State Operation of Magnetic Fusion Devices

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IAEA Headquarters, Vienna, Austria

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Topics

I. Superconducting Devices

II. Long-Pulse operation and Advanced Tokamak Physics

III. Steady State Fusion Technologies and Plasma Wall Interactions

IV. Long Pulse Heating and Current Drive

V. Particle Control and Power Exhaust

VI. ITER-Related Research and Development Issues

Schedule

Monday, 20th March

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10.45-11.30	**I-1:** S. Mirnov *Evolution of SS tokamaks and their operational limits*
11.30-11.40	**Coffee Break**
11.40-12.25	**I-2:** M. Gryaznevich *Advancing Research Towards Steady-state operations: ST40 project*
12.25-13.10	**I-3:** B. Kuteev *Development of DEMO-FNS Steady State Tokamak for Hybrid Technologies*
13.10-14.40	**Lunch Break**
Session 2. Chair: S. Kubo	
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15.40-16.10	**O-3:** H. Ohwada *Investigation of plasma wall interaction based on plasma radiation measurement in long duration discharge on QUEST*
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16.20-16.50	**O-4:** G. Stankunas *Activity inventories and decay heat of ITER material samples after long-term irradiations with 14 MeV fusion neutrons at JET*
16.50-17.20	**O-5:** A. Vertkov *Experience in the development of liquid metal plasma facing elements based on capillary pore structure for steady state operating tokamak*
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11.10-11.55	**I-5:** S.-H. Hahn *Access to long-pulse high-beta operation regime in KSTAR*
11.55-12.40	**I-6:** Z.P. Luo *Steady-state ELM-free H-mode Quasi-Snwoflake discharge in EAST*
12.40-14.10	**Lunch Break**
Session 4. Chair: B. Kuteev	
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14.40-15.10	**O-8:** J. Chung *Formation of the internal transport barrier in KSTAR*
15.10-15.40	**O-9:** J. McClenaghan *Transport modeling of the DIII-D high βp scenario and extrapolations to ITER steady-state operation*
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15.50-16.20	**O-10:** X.M. Song *Exotic Plasma Shape on HL-2M*
16.20-16.50	**O-11:** J. Hillairet *Material, Radio-Frequency and Mechanical Characterisations of High Current Steady-State Sliding Contacts for the ITER ICRH Antenna*
16.20-16.50	**O-12:** C. Hopf *Technological considerations for the steady state operation of an NBI beamline for heating and current drive*
16.50-17.20	**O-13:** D. Rittich *Quantitative access to Neutral Beam Current Drive Experiments on ASDEX Upgrade*
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Session 5. Chair: X. Litaudon	
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10.15-11.00	**I-7:** J. Qian *Progress on the development and physics basis of high betaP scenario on EAST and DIII-D for ITER steady state operation*
11.00-11.10	**Coffee Break**
11.10-11.55	**I-8:** Y.-K. Oh *Status of the high performance and long pulse operation in KSTAR and exploring the issues in ITER and K-DEMO*
11.55-12.40	**I-9:** S. Kubo *RF Heating System Development for Steady State Operation in the Large Helical Device*
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14.10-16.10	**SSOCG** **Chair: A.C.C. Sips**
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Session 6. Chair: S. Chaturvedi	
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10.15-11.00	**I-11:** R.J. Goldston *New Scaling for Plasma Detachment and its Implications for Tokamak Power Plant Design*
11.00-11.30	**O-14:** G.M. Wallace *High Field Side Lower Hybrid Launch for Steady State Plasma Sustainment*
11.30-11.40	**Coffee Break**
11.40-12.10	**O-15:** K. Gałązka *Impurity seeding in JT-60SA carbon divertor configuration for efficient power exhaust in steady state discharges* J. Chung *Formation of the internal transport barrier in KSTAR*
12.10-12.40	**O-16:** M. Kozulia *Continuous wall conditioning Continuous wall conditioning field using VHF discharge*
12.40-13.10	**O-17:** D. Moseev *ECH Experiments and Stray Radiation Study in Wendelstein 7-X in the Context of Steady-State Operation of Fusion Devices*
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14.30-16.00	**Summary and Closing**
16.00	**End of 9th IAEA TM on Steady State Operation of Magnetic Fusion Devices**

Abstracts

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OV-1: H.-S. Bosch, *First results of the operation phase OP 1.1 of Wendelstein 7-X*

OV-2: J. Bucalossi, *Start of the WEST facility*

OV-3: S. Ide, *The Advances in the assembly of JT-60SA and the research plan*

OV-1: First Results of the Operation Phase OP 1.1 of Wendelstein 7-X

H.-S. Bosch for the W7-X Team

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The main objective of the Wendelstein 7-X stellarator (W7-X) is to demonstrate the integrated reactor potential of the optimized stellarator line. An important element of this mission is the achievement of high heating-power and high confinement in steady-state operation. The approach to this mission is following three steps. First plasmas were produced in a limiter configuration (OP 1.1), then a test divertor unit is being installed (TDU) for the next campaign, OP 1.2, before the full steady state capability will be achieved implementing active cooling of all in-vessel components and a steady state high heat flux divertor. This talk presents the preparation of OP 1.1, the first results from OP 1.1 and the further completion of W7-X and the preparation of OP 1.2.

In 2014/15, after the closure of the outer vessel, the commissioning of the W7-X device was started, while the installation of the in-vessel components, diagnostic systems and peripheral components was continued. After the evacuation of the cryostat vessel and the plasma vessel, and checks of the mechanical stability of the vessels, the leak-search and cleaning of the 2000 m cryo-piping was started. In spring 2015 the magnetic coil set together with the support system was cooled down to 4 K. In the next step, the superconducting magnet system was loaded with currents for the first time. After integral commissioning of the magnet system, magnetic flux surfaces were confirmed using an electron beam. Subsequently, the plasma vessel was baked to 150° C, and the central safety system was commissioned and validated.

In December 2015, the first helium plasma was generated using ECRH, in February 2016 the working gas was switched to hydrogen. The first operation phase (OP 1.1) was successfully finished in March 2016. At the end of OP 1.1 the discharge duration was close to 6 seconds, the limit for the heating energy was increased to 4 MJ and electron temperatures of ~10 keV were achieved. Due to the low densities in the range of 10^{19} cm^{-3} and the pure electron heating by ECRH, the ion temperatures reached only 2 keV.

At present, W7-X is undergoing the next completion phase, including the installation of the test divertor unit, the installation of the carbon tiles on the inner plasma vessel wall, an upgrade of existing diagnostics and the installation of new diagnostics.

OV-2: Start of the WEST Facility

J. Bucalossi[1], M. Missirlian[1], P. Moreau[1], F. Samaille[1], E. Tsitrone[1], T. Alarcon[1], L. Allegretti[1], S. Antusch[2], T. Batal[1], O. Baulaigue[1], F. Bouquey[1], C. Bourdelle[1], S. Bremond[1], C. Brun[1], B. Cantone[1], M. Chantant[1], C. Chavda[3], J. Colnel[1], E. Corbel[1], Y. Corre[1], X. Courtois[1], R. Dejarnac[4], E. Delmas[1], L. Delpech[1], C. Desgranges[1], P. Devynck[1], L. Doceul[1], D. Douai[1], H. Dougnac[1], K. Ezato[5], F. Faïsse[1], P. Fejoz[1], F. Ferlay[1], M. Firdaouss[1], L. Gargiulo[1], S. Gazzotti[1], S. Gharafi[1], J.C. Giacalone[1], C. Gil[1], H. Greuner[6], E. Grigore[7], D. Guilhem[1], J. Gunn[1], A. Hakola[8], J.C. Hatchressian[1], P. Hennequin[9], M. Houry[1], C. Klepper[10], S. Larroque[1], T. Loarer[1], P. Lotte[1], G.-N. Luo[11], C. Martin[12], A. Martinez[1], O. Meyer[1], P. Mollard[1], E. Nardon[1], R. Nouailletas[1], G. Pinsuk[13], G. Raupp[6], N. Ravenel[1], C. Reux[1], M. Richou[1], H. Roche[1], C. Ruset[7], F. Sabathier[1], R. Sabot[1], A. Saille[1], F. Saint-Laurent[1], A. Santagiustina[1], B. Santraine[1], Y. Seki[5], D. Sestak[4], S. Suzuki[5], D. Thouvenin[1], J.M. Travère[1], J.M. Verger[1], L. Vermare[9], Y. Wang[14], A. Werner[15], B. Zago[1], and the WEST team

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After 4 years of construction, the WEST facility, which is an in-depth evolution of the superconducting tokamak Tore Supra into a test bench for ITER divertor plasma facing unit (PFU), has just started its operation. First plasma breakdowns were achieved in December 2016 while the new in-vessel divertor coils were qualified at their nominal current. Plasma commissioning is now in progress. The first PFUs prototypes featuring full-scale ITER tungsten monoblock technology and manufactured by 3 different suppliers have been integrated in the lower divertor target. A progressive power loading using LHCD and ICRF heating systems up to ITER relevant steady state peak heat loads is planned. In this first phase of operation, pulse duration will be limited by the inertially cooled startup elements that complete the lower divertor target. Besides, the upper divertor target is already fully equipped with actively cooled tungsten armoured elements and will allow for preparing long pulse operation in upper null configuration. The paper will describe the capabilities of this new steady-state device, the status of its commissioning and the foreseen experimental program.

OV-3: The Advances in the Assembly of JT-60SA and the Research Plan

S. Ide[1] and the JT-60SA team

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The JT-60SA (Super Advanced) project is in progress under the Broader Approach Satellite Tokamak Programme jointly implemented by Japan and Europe, and under the Japanese national program. The objective of the project is to support researches on ITER and develop steady state high beta physics and engineering basis towards DEMO reactor. [1]. Manufacturing of tokamak components has progressed well. Accuracy of the components has been found to be as high or even higher as designed. For example, the circularity (deviation from the true circle) of the EF (equilibrium field) coils is a fraction of designed value, less than 1 mm for five out of six coils. Having manufacturing of components in progress and many of them arrived on site, the JT-60SA construction has engaged into the assembly phase on the site from the manufacturing phase. As of the end of January, five TF coils have arrived on site, and two of them have been already installed on their position. As the assembly proceeds, accuracy of the assembling components has also been fund to be good. In this talk, the latest status of the assembly with emphasis on its accuracy and how the accuracy has been achieved will be reported. In parallel to the engineering works, research activities in preparation for the JT-60SA experiment have been underway in wide collaboration between Japanese and EU researchers. Modeling and simulation on various issues, from the plasma core to the SOL/divertor domain, has been carried out on both sides and compared each other for more reliable prediction of the plasma performance and scenario development. The JT-60SA Research Plan has been published and updated periodically, the latest version is 3.3 [2]. Status of the research collaboration in preparation for the JT-60SA experiments will be reported as well.

References:

[1] SHIRAI, H., et al., OV/3-3 Fusion Energy Conference, Kyoto 2016.

[2] JT-60SA Research Plan v3.3, http://www.jt60sa.org/pdfs/JT-60SA_Res_Plan.pdf.

List of Invited Orals:

I-1: S. Mirnov, *Evolution of SS tokamaks and their operational limits*

I-2: M. Gryaznevich, *Advancing research towards steady-state operations: ST40 project*

I-3: B. Kuteev, *Development of DEMO-FNS steady state tokamak for hybrid technologies*

I-4: A. Bock, *Non-Inductive improved H-mode operation in ASDEX upgrade*

I-5: S.-H. Hahn, *Access to long-pulse high-beta operation regime in KSTAR*

I-6: Z.P. Luo, *Steady-state ELM-free H-mode Quasi-Snwoflake discharge Ain EAST*

I-7: J. Qian, *Progress on the development and physics basis of high betaP scenario on EAST and DIII-D for ITER steady state operation*

I-8: Y.-K. Oh, *Status of the high performance and long pulse operation in KSTAR and exploring the issues in ITER and K-DEMO*

I-9: S. Kubo, *RF heating system development for steady state operation in the large helical device*

I-10: M. Endler, *Toward steady-state divertor operation of Wendelstein 7-X*

I-11: R. J. Goldston, *New scaling for plasma detachment and its implications for tokamak power plant design*

I-1: Evolution of SS Tokamaks and Their Operational Limits

S.V. Mirnov

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55 years ago, during the winter 1961-62YY E.P. Gorbunov and K.A. Razumova had obtained [1] the macroscopically stable discharge with the duration a few milliseconds and with the plasma temperature up to 100eV on the new tokamak TM-2 (a small tokamak-2) in Division of Plasma Research (DPR) of the Kurchatov Institute in Moscow. This has seemed absolutely unusual for the practices of Plasma Research. The initial doubts of physicists about the runaway electron nature of these strange discharges turned out unfounded. One year later, a similar macroscopically stable discharge was obtained by L.A. Artsimovich et al on the largest for those times T-3 tokamak. It has been found that the existence of this type of discharge is substantially longer than the characteristic energy lifetime of the plasma. That means it had quasi-stationary character. Thus, the active investigations of a new tokamak phenomenon have begun. Finally, today we know the basic principles and criteria for the existence of such type quasi-stationary discharge on the ground of these investigations. One of these seems most important for the steady state tokamak. The comparative analysis of so-called "high performance tokamak regimes" shows a one important feature. During the all 55 years operations the heating power ***P*** allowable for such regimes changes approximately linearly with the ***S*** – the area of the tokamak chamber facing to the plasma [2]. That means the relation ***P/S*** seems as some kind invariant for conventional short pulse tokamaks operated during 0.01 – 10sec. However, already first operations of superconducting tokamaks (TRIAM, Tore Supra, for example) discovered that this parameter should be progressively reduce if you need increase the tokamak pulse duration Δt ($\Delta t \sim 1/P^{1.7}$). The electrical breakdown of sheath on the plasma-wall boundary can be suggested as the simplest explanation of this phenomenon [3]. Its probability will be increased as increase the thickness of deposition films of erosion products of limiters and chamber walls facing to the plasma. That means the steady state plasma process, which is necessary for the transition from laboratory devices to fusion reactor, will require the online removal the erosion products from the tokamak chamber. The use of lithium circulating in the form of a closed loop between the wall and plasma promises a solution of this problem. In communication it will be discussed the last experimental achievements in this field.

References:

[1] GORBUNOV, E.P., RAZUMOVA, K.A.,1963 *Atomic energy* **15 Rus.** №5 363-370;
[2] MIRNOV, S.V. Plasma Phys. Contr. Fus. **55** (2013) 045003;
[3] MIRNOV, S.V. Plasma Phys. Contr. Fus. **58** (2016) 022001.

I-2: Advancing Research Towards Steady-State Operations: ST40 project

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To advance research towards steady-state operations at burning plasma conditions, Tokamak Energy is performing R&D in several areas:

- High field compact ST40 (3T/2MA) is under construction to demonstrate advantages of high field STs, i.e. high **bootstrap** fraction (due to high beta and high elongation), which for an economic ST reactor should be **above 90%**, combined with (supported by) high plasma pressure**.**
- **HTS magnets** – the best candidate for magnets in steady-state devices due to high J_{crit} at B > 20T. Several prototypes are under development by TE.
- Resolving issues in connection with steady-state DT operations – all range from **regulations** and **safety** to **tritium plant** for a low-power tritium experiment, and, of course, issues connected with **physics of steady-state** regimes and control in a **burning plasma** devices. TE is considering to conduct limited DT operations in ST40 to demonstrate the performance of high field STs in these regimes. The next step TE device will certainly operate in DT and so these issues require urgent advances.

Progress in these areas will be reported, including update on the status of construction of ST40.Advances in the development of high temperature superconductors (HTS) [1], successful demonstration of >24h discharges in ST25-HTS tokamak with all-HTS YBCO magnets [2] and encouraging results on a strong favourable dependence of electron transport on higher toroidal field (TF) in Spherical Tokamaks (ST) [3,4] open new prospects for a high field ST as a very compact fusion reactor. The combination of the high β (ratio of the plasma pressure to magnetic pressure), which has been achieved in STs [5], and the high TF that can be produced by HTS TF magnets, opens a path to lower-volume fusion reactors, in accordance with the fusion power scaling proportional to $\beta^2 B_t^4 V$. However, technical challenges connected with application of **HTS** (cables and demountable joints, stresses & strain, heating due to neutrons, damage under irradiation and so shielding optimisation) require intensive R&D, which we are performing, and **intermediate results will be presented**. Also, steady state operations in a burning plasma device may have specific issues, different to those in present tokamaks and these issues will be discussed in detail.

The demonstration of reliable steady state operations in a compact high field ST, even at the level of a few MW fusion output as a first step, will significantly advance not only fusion research but also the commercial exploitation of fusion and will be an important step towards the development of industrial fusion power production. Discussions are under way in TE as to how ambitious the next step device should be and a requirements document is in preparation.

References:

[1] GRYAZNEVICH, M., *et al*, “Progress in applications of HTS in Tokamak Magnets” 2013 *Fus. Eng. & Des.* **88** 1593.
[2] GRYAZNEVICH, M., *et al*, 2014 24th IAEA FEC St Petersburg 13-18 October 2014 OV/P-04.
[3] VALOVIC, M., *et al*, 2009 *Nucl. Fusion* **49** 075016.
[4] KAYE, S., *et al*, 2007 *Nucl. Fusion*, **47** 499.
[5] GRYAZNEVICH, M., *et al*, 1998) *Phys. Rev. Lett.* **50** 3972.

I-3: Development of DEMO-FNS Steady State Tokamak for Hybrid Technologies

B. Kuteev and DEMO-FNS team

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Fusion-fission hybrid facility based on superconducting tokamak DEMO-FNS is developed in Russia for integrated commissioning of steady-state and nuclear fusion technologies at the power level up to 40 MW for fusion and 400 MW for fission reactions. Fusion-fission hybrid technologies tested on DEMO-FNS may accelerate implementation of fusion technologies and are capable to improve the neutron balance in the global nuclear energy system. Implementation of hybrid technologies should also accelerate the development of Atomic Energy reducing the radio-toxicity generated in nuclear fuel cycle and the level of pollution by fuel reprocessing. These problems become actually at the forthcoming transition of Nuclear Power to a closed nuclear fuel cycle.

The development of fusion neutron source DEMO-FNS based on classical tokamak was launched in the NRC "Kurchatov Institute" in 2013. Design was aimed at reaching steady state operation of the plant with a neutron load ~ 0.2 MW/m^2, lifetime neutron fluence ~ 2 MWa/m^2, with the plasma facing surface area of the blanket ~ 100 m^2, sufficient for testing materials and components in the fusion neutron spectrum as well as for development of hybrid transmutation technology, fuel nuclides and tritium production, and energy generation.

Significant factors of changing the initial design are the necessity to increase the radiation shield thickness and strength characteristics of the electromagnetic system. An increase of large radius from 2.5-2.7 m to 3.2 m changed plasma parameters. However, simulations showed a weak influence of geometry on the DT fusion power and full plasma current. The results of three-dimensional modeling of neutron flux from the plasma and blanket sources will be discussed in the report. It was shown that the hybrid blanket adds a small contribution to the radiative heating of the toroidal field coils. Analyses of the interaction of DEMO-FNS facility with the nuclear fuel cycle of Russia's nuclear power industry will be addressed as well.

Design parameters of a tokamak design for DEMO-FNS are the following.

Aspect ratio R/a,m	3.2/1 m
Toroidal magnetic field	5 T
Electron/ion temperature $T_e(0)/T_i(0)$	1.5/10.7 keV
Beta normalized β_N	2.1
Beta poloidal β_p	0.96
Plasma current I_{pl},	4.5 MA
Neutron yield G_N	$1.3\cdot10^{19}$/s
Neutral injection power	36 MW
ECR heating power	6 MW
Discharge time	up to 5000 hours
Capacity factor	0.3
Life time	30 years
Consumed/generated power	up to 200 MW

I-4: Non-Inductive Improved H-mode Operation in ASDEX Upgrade

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M. Bernert, A. Burckhart, M. Dunne, B. Geiger, L. Giannone, V. Igochine, A. Kappatou,
R. McDermott, A. Mlynek, T. Odstrˇcil, G. Tardini, H. Zohm, and the ASDEX Upgrade Team

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Conventional tokamak high-confinement mode (H-mode) scenarios are susceptible to magnetohydrodynamic (MHD) instabilities and also depend on inductive current from the central solenoid to maintain the plasma current. Raising the central q-profile with electron-cyclotron current drive (ECCD) and neutral beam current drive (NBCD) can not only improve the stability and confinement of the discharge by eliminating some of the most common resistive MHD instabilities, but also increase the pulse length by increasing the core bootstrap current density $j_{bs} \sim q\nabla p$. The corresponding broadening of the current profile further elevates the central q-profile, which allows the resulting current density profile to maintain itself but also reduces the ideal stability limit.

This contribution presents recent experimental results from full-metal ASDEX Upgrade in which non-inductive operation at I_p = 800 kA with q_{95} = 5.3 and β_N = 2.7 was achieved using off-axis ECCD (~ 10%), off-axis NBCD (~ 40%) and bootstrap current ($\gtrsim$ 40%). The scenario is marked by an initial external manipulation of a relaxed current profile via ECCD and NBCD, followed by a feedback-controlled increase of NBI heating to reach and finally maintain the target-β in stationary conditions. Operation without induction by the central solenoid over about one resistive time τ_R maintains more than 98.5% of I_p. For $\beta_\mathrm{N} \gtrsim 2.7$ the scenario suffers from ideal MHD modes. Consequently, since β must be maximised for high fusion performance and bootstrap current generation, future investigations will aim at extending the ideal stability limit.

Furthermore, to verify the theoretical understanding of plasma transport in the non-inductive phase ($H_{98}(y,2) > 1.1$), the kinetic profiles of the discharges are modelled with TGLF and compared to the experimentally observed ones. The results of this comparison suggest that TGLF is significantly over-estimating ion heat transport in the presence of high β and/or high fast ion density.

Finally, since an accurate determination of the q-profile is a prerequisite for such advanced scenario studies, this contribution will report on the presence of polarised background light that contaminates the measurements of the Motional Stark Effect diagnostic.

I-5: Access to Long-Pulse High-Beta Operation Regime in KSTAR

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Addressing scientific and technical issues on extending high performance scenarios to a long pulse, steady-state condition is essential research for the realization of sustainable economic fusion reactors. In this paper, results of recent experimental approaches are presented on achieving stationary high-beta in long-pulse operations in KSTAR, which has full superconducting toroidal & poloidal coils most similar as ITER. The research has focused on achieving high-β_N and high-βp scenarios under present available external heating power P_{ext} ~ 6MW and the pulse length allowed by the present plasma facing component (PFC). Conditions of maximizing each parameter have been investigated mainly by exclusive parametric scans of the principal knobs, such as toroidal magnetic field (Bt), beam injection power (P_{NB}), and plasma current (Ip). For high-β_N, it is possible to create a β_N >3 plasma for 3 seconds by tailoring the Ip ramp and the pulse design of beam injection, and the limitation of the length of high performance comes from the MHD activities and the corresponding confinement degradation [1]. For high- βp, a 20s fully-noninductive discharge with βp =2.7, β_N =2.0 is experimentally demonstrated at Ip = 0.4 MA, Bt=2.5T, Pext = 4.7MW. The control of the density, the outer gap, striking points, and the fast-ions losses is essential to reduce the temperature increase of outboard limiters so that the plasma can be maintain a favorable particle balance within the allowed engineering limits on the PFC. Learned techniques have been utilized to successfully achieve a 71s H-mode discharge at βp = 1.9~2.4, Ip = 0.45MA with Bt= 2.5 T.

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[1] YOON, S.W. et al., proceedings in: 26th IAEA Fusion Energy Conference, Kyoto, Japan, Oct 16-22 (2016) EX/P4-1.

This work was supported by Ministry of Science, ICT, and Future Planning under KSTAR project.

I-6: Steady-State ELM-Free H-mode Quasi-Snwoflake Discharge in EAST

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High-performance and steady-state operation is a crucial goal of EAST and other magnetic fusion devices [1]. A great challenge facing future fusion devices is to achieve a steady-state H-mode plasma regime with acceptable divertor heat fluxes. Advanced magnetic divertor configuration is one of the attractive methods to spread the heat fluxes over divertor targets in tokamak because of enhanced scrape-off layer transport and an increased plasma wetted area on divertor target. Exact snowflake (SF) 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) or X-divertor configuration, characterized by two first-order nulls with primary null inside and secondary null outside the vacuum vessel. Both modeling and experiment showed this QSF can result in significant heat load reduction to divertor target [2]. In order to explore the plasma operation margin and effective heat load reduction under various plasma conditions and QSF shape parameters, we developed ISOFLUX/PEFIT shape feedback control. In experiment, we firstly applied the control of QSF in a similar way to control the single null divertor configuration, with specially designed control gains. Reproducible QSF discharges have been obtained with stable and accurate plasma boundary control. With advanced Li wall conditioning and current drive and heating capabilities of low hybrid current drive (LHCD), ion cyclotron resonance wave (ICRH) and electron cyclotron resonance wave (ECRH), EAST have achieved highly reproducible steady-state ELM-free H-mode QSF discharge, with the pulse length up to 20s, about 450 times the energy confinement time. This is the first time for EAST to achieve the steady-state H-mode operation with QSF configuration which is ELM-free and yet has good density control. The capability of the QSF to reduce the heat loads on the divertor targets has been confirmed. This new QSF H-mode regime may open a new avenue for ITER and CFETR fusion development.

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[2] CALABRO, G., et al, Nucl. Fusion 55 (2015) 083005.

I-7: Progress on the development and Physics Basis of High betaP Scenario on EAST and DIII-D for ITER Steady State Operation

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Recent high betaP experiments on EAST and DIII-D have successfully developed plasmas with the feature of good energy confinement at low rotation, an important contribution to ITER steady state scenario research. On EAST, fully non-inductive steady state H-mode plasma (H_{98y2} > 1.1) has been extended over 60s for the first time with ITER-like W divertor by sole RF heating and current drive. Meanwhile, ELM suppression has also been achieved with n = 1 RMP for a long pulse H-mode discharge with conditions similar to the long pulse H-mode. More recently, higher confinement (H_{98y2} > 1.4) with V_{loop} ~ 0.0V was obtained when the scenario is extended to higher betaP ~2.0 by RF heating together with NBI. Preliminary results show that stationary electron ITB results in this good confinement. On DIII-D, extension of large radius ITB, high q_{min}, high bootstrap current fraction scenario toward low plasma rotation and q_{95} relevant to ITER steady state operation has been successfully demonstrated. The experiments have shown that the key feature of large radius ITB, which results in excellent energy confinement characteristics, is maintained when the scenario is extended inductively to higher plasma current, for lower q95, and more balanced NBI, for lower plasma rotation. Transport analysis suggests that ExB shear has little contribution to turbulence suppression, while Shafranov shift has the key stabilizing effect on turbulence.

Together, these results provide strong encouragement that high performance, high betaP regimes could be utilized for ITER steady state operation with good energy confinement at low plasma rotation.

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I-8: Status of the High Performance and Long Pulse Operation in KSTAR and exploring the Issues in ITER and K-DEMO

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Korea Superconducting Tokamak Advanced Research (KSTAR) program has mission to explore the scientific and technical issues in steady-state high performance plasma achievement that are essential to ITER and Korea demonstration reactor, K-DEMO. In this regards, KSTAR has made a remarkable progress in developing long pulse and high performance operation. The outstanding engineering features of KSTAR, such as extremely low intrinsic error field, low toroidal field ripple, flexible in-vessel control coil, and advanced 2D/3D imaging diagnostics systems, enable the exploration of the plasma confinement and instability under extremely advanced operation window.

According to improvement in the plasma control, KSTAR could achieve an extremely long pulse H-mode discharge up to 70s in 2016. In addition to long pulse H-mode operation, various operation modes have been explored such as sustaining the internal transport barrier formation in electron and ion profile up to 7s, sustaining the high normalized beta scenario (β_N >3) up to 3s, achieving large fusion gain (G ~ 0.38) in hybrid mode, and achieving very low edge safety factor (q95 < 2.3) in H-mode. As a high priory research topic in ITER baseline operation, KSTAR has investigated a very robust and reproducible edge localized mode (ELM) crash suppression window in n=1 by adopting appropriate plasma model.

In 2017, KSTAR will be operated with increased PFC temperature up to 150 degree of Celsius, liquid helium circulation in in-vessel cryopump, and operation of a pellet injector for the improved particle control. And in-depth exploration of advanced operation scenarios and long pulse operation will be conducted by the improved diagnostics and analysis.

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I-9: RF Heating System Development for Steady State Operation in the Large Helical Device

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The steady state operation (SSO) of high-performance plasma in LHD has advanced based on the improvements and upgrading of RF heating systems. Here, RF system includes, ion cyclotron range of frequency (ICRF) and electron cyclotron resonance heating (ECRH). The heating power of steady state ICRF and ECRH exceeded 1 MW and 500 kW, respectively, and the higher-density helium plasma with minority hydrogen ions was maintained by using the HAS antenna and new 77 GHz/154 GHz gyrotrons. As a result, plasma performance improved; e.g., an electron temperature of more than 2 keV at a density of more than 2×10^{19} m^{-3} became possible for more than 1 min. HAS antenna that can control the launched parallel wave number and heated a core plasma efficiently have been installed, and enabled the analysis of the heat flow balance and particle flux balance of steady-state operation. Particle balance analysis indicates that externally fed helium and hydrogen particles are mainly absorbed by a chamber wall and divertor plates, even after the 54-min operation. The improvement and upgrading of the ECRH system includes several improvements in the transmission line, and introduction of high power 77 and 154 GHz gyrotrons. A booster ECRH power of 400 kW in 300 ms pulse and/or ICRF power is also prepared and succeeded to avoid the radiation collapse, apart from the CW power. One of the main reasons of the limitation of the discharge duration has been the sudden termination due to the increase in the impurity influx, which is attributed to be the metallic flake from the wall. This impurity influx can induce radiation collapse when the injection power mis critical to maintain the discharge. Eventually, the booster injection is triggered by a sudden increase of the electron density and demonstrated that collapsing discharge successfully revived in some cases.

The details of these improvement and upgrading of the RF systems and development of the termination avoidance scenario related to the RF heating are discussed.

I-10: Toward Steady-State Divertor Operation of Wendelstein 7-X

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The magnetic field structure of the Wendelstein 7-X (W7-X) stellarator inherently offers a divertor geometry. This is formed by a chain of natural magnetic islands residing at flux surfaces with low-order rational values of the rotational transform. The engineering task was to design the target plates such that the islands are intersected far from the plasma centre, that they can sustain the resulting load of 10MW heating power under stationary conditions, and that they are compatible with different magnetic configurations, permitted by the versatile magnet system of W7-X.

This is achieved by an open divertor which consists of ten units, following the fivefold toroidal and the up-down stellarator symmetry of the device. The divertor units are located along the narrow sides of the elongated plasma cross-section, where the geometric distance between plasma centre and plasma edge is maximum. The surface of the target plates is adapted to the shape of the flux surfaces, such that the magnetic field lines meet the target surface under angles below 3° in the areas with highest heat flux, thus limiting the effective power load to the targets to the specified value of $10MW/m^2$.

The High Heat Flux (HHF) divertor constructed for steady-state operation of W7-X consists of individual target elements, each of which is made from CFC tiles bonded to a water-cooled CuCrZr heat sink. Extensive tests were performed to demonstrate that these target elements satisfy the thermal load requirements. Although this design is capable to take the specified steady-state power loads, it is sensitive to thermal overloading, which might occur due to toroidal or up-down asymmetries. Infrared thermography will therefore be employed for a continuous surveillance. Before installation of this HHF divertor, W7-X will be operated for approximately one year with an adiabatically loaded Test Divertor Unit (TDU), which is built from thicker fine grain graphite elements with the same surface shape as the HHF divertor. In this operation phase (OP1.2), the energy per plasma pulse will be limited to ∼80MJ (e. g., 10 s at 8MW heating power), and models of divertor heat load patterns can be verified on this temporary structure before installing the HHF divertor.

Beyond the measurement of asymmetries in the divertor heat load patterns, this operation phase will serve for a first exploration of divertor physics. Unlike a tokamak divertor, the connection lengths parallel to the magnetic field between the stagnation point and the target can reach several 100 m in the island divertor of a low shear device, in spite of the short geometric distance between last closed magnetic surface and target. The mechanism which can cause plasma detachment in an island divertor will be discussed. Furthermore, the investigation of particle exhaust and density control will constitute important parts of the OP1.2 divertor physics programme.

I-11: A New Scaling for Plasma Detachment and its Implications for Tokamak Power Plant Design

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ITER and future reactors depend on divertor "detachment," whether partial, pronounced or complete, to limit heat flux to plasma-facing components and to limit surface erosion due to sputtering. It would be valuable to have a measure of the difficulty of achieving detachment as a function of machine parameters, such as input power, magnetic field, major radius, etc. Frequently the parallel heat flux, estimated typically as proportional to P_{sep}/R_0 or $P_{sep}B_0/R_0$, is used as a proxy for this difficulty. Here we argue that impurity cooling is dependent on the upstream density, which itself must be limited by a Greenwald-like scaling. Taking this into account self-consistently, we find the impurity fraction, c_z, required for detachment scales as:

$$c_z \propto \frac{P_{sep}}{\langle B_p \rangle \left(1+\kappa^2\right)^{3/2} f_{GW,sep}^2 \ell_{\|}^*} \left(\frac{1+\bar{Z}}{\bar{A}}\right)^{1/2}$$

where $\ell_{||}^*$ represents enhancement in the divertor connection length due to advanced divertor configurations. The absence of any explicit scaling with machine size is concerning, as P_{sep} surely must increase greatly for an economic fusion system, while increases in the other parameters are limited. This result should be challenged by comparison with 2-D codes and measurements on existing experiments. Nonetheless, it suggests that higher magnetic field, stronger shaping, double-null operation, "advanced" divertor configurations, as well as alternate means to handle heat flux such as metallic liquid and/or vapor targets merit greater attention. In particular, lithium appears to be an attractive radiator in this context.

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O-2: J.-G. Kwak**,** *Long term renovation inside KSTAR vacuum vessel toward steady state operation*

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O-16: M. Kozulia, *Continuous wall conditioning in a toroidal device without magnetic field using VHF discharge*

O-17: D. Moseev**,** *ECH Experiments and stray radiation study in Wendelstein 7-X in the context of steady-state operation of fusion devices*

O-1: Physical Changes Study of the Materials Exposure to Fusion Grade Plasma in a Low Energy Plasma Focus Device

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One of the key issues still to be resolved in the quest for fusion energy production is the characterization and qualification of candidate materials for plasma facing components (PFC). Hence, the PFCs are directly exposed to extreme radiation and heat load conditions in fusion reactors, they must work in extreme adverse environment. Many different materials and their alloys/compounds are under consideration for PFCs which include tungsten, beryllium, molybdenum, graphite, boron carbide, carbon fiber composite and stainless steel. This paper aims to investigate the effects of fusion relevant intense pulse of energetic and high fluence deuterium ions and neutrons generated in a low energy (3.0 kJ) plasma focus device (UNU/ICTP PFF) on stainless steel-AISI 304 and tungsten samples. Characterization of irradiated samples is performed using X-ray diffraction (XRD), field emission scanning electron microscopy (FESEM), energy dispersive X-ray (EDX) spectroscopy, X-ray photoelectron spectroscopy (XPS), atomic force microscopy (AFM) and micro hardness tester. The unique characteristics of the plasma focus device, the use of ferritic-martensitic steel as the main structural material for the ITER vacuum vessel and in vessel components and the utilization of tungsten as the most important material for first wall of fusion reactor in the diverter and baffle regions, motivated us to investigate the combined effect of energetic ions and fusion neutrons emanated from deuterium plasma on the structural, morphological and hardness of the irradiated samples.

O-2: Long Term Renovation inside KSTAR Vacuum Vessel toward Steady State Operation

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Fusion reactor needs the steady state operation and fully non-inductive current drive is essentially one of the top priority issues in tokamak operation. It is believed that most of CD should be covered by bootstrap in tokamak reactor but still the development of the efficient external CD sources is needed for the precise profile control of CD in advanced tokamak scenario. Recently it is known that the top or mid plane launch of RF is very efficient including high field side launch.

KSTAR has a lot of unused space inside the vessel which is now used for the passive stabilizer and water piping to PFC components. Original concept using the low ripple effect of toroidal field made the low field side of the wide space not to be used. So the basic idea of long term upgrade plan of KSTAR is by pushing out the present passive stabilizer to the outer vessel wall as close as possible, we can secure more wide space for plasma volume. The difficulty of higher ripple at outer low field side could be overcome by inserting ferritic steel to the vessel wall. By using the lower single null configuration with available outer space, it is possible to install the top launch ECH and off plane LHCD and Helicon through upper vertical port.

In addition, suggested upgrade plan is expected to provide more space for exploring the new diverter configuration by installing additional shaping coil and diagnostics for diverter study. Including above topics, the presentation will address the recent results and engineering issues on KSTAR toward long pulse steady state operation.

O-3: Investigation of Plasma Wall Interaction Based on Plasma Radiation Measurement in Long Duration Discharge on QUEST

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The long duration discharge was sustained for 2 hours by controlling the gas balance with the hot-wall system installed in the QUEST vessel [1]. The plasma shaping and boundary were monitored by camera. Visible light measured by camera at the upper side was brighter than that at lower side. It seems that the asymmetry of light between the upper (direction of ion toroidal drift) and lower (direction of electron toroidal drift) side became strong in longer duration discharge. To confirm the asymmetric property quantitatively, the fiber spectrometer system was considered and developed. The fiber line of sight was mechanically scanned in the poloidal cross section to observe the asymmetric property.

As a result of measuring for 2 hours' continuous discharge which was controlled by Ha feedback system (Ha feedback system measures Ha and control fuel injection rate for constant density during long discharge.), the Ha intensity of center direction increase to twice, and upper direction increase to 3 times from early period of discharge. Moreover, the ratio of center side to upper side was almost 1.5 times at early period, but at the later period it became 1. That is, even though the overall neutral particle density rose, the increase of the neutral particle density of the upper side was superior than the other side.

In order to understand asymmetric property, we observed the surface of the vessel wall. As the result, the deposition layer of the upper side is localized depend on the incident direction of the ion. On the other hand, lower side deposition layer is formed uniformly thick. Therefore, in order to change the incident direction of ion, we experimented by changing the direction of plasma current. As the result, the emission line of Carbon (229nm) rose 1.5 times on upper side. On the other hand, lower side carbon intensity decreased slightly. It suggests that the deposition layer on upper side wall was sputtered by injection ion which direction changed. Moreover, the total amount of fuel gas was controlled so that the constant plasma density was not different from that before the changing the direction of plasma current. It means that the wall retention increases, and it suggests that the macro wall gas exhaust increases since hydrogen enters thick deposition layer part.

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O-4: Activity Inventories and Decay Heat of ITER Material Samples after Long-Term Irradiations with 14 MeV Fusion Neutrons at JET

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Irradiations with 14 MeV fusion neutrons are planned at JET in DT operations with the objective to validate the activation of structural materials and the radiation damage in functional materials expected in ITER and fusion plants. This study describes the activation, dose rate and radiation damage calculations performed for materials irradiated during the complex and long-term irradiation sequence for DD+HH+TT+HH+DD phases, which was taken into consideration where the samples of ITER sample materials are irradiated. The materials included those used in the manufacturing of the main in-vessel components, such as ITER-grade W, Be, CuCrZr, 316L(N), and the functional materials used in diagnostics and heating systems. The neutron induced activities and dose rates at shutdown were calculated by the FISPACT code, using the neutron fluxes and spectra that were provided by the preceding MCNP neutron transport calculations. The study of investigated types of steels identified the Mn-56 and Cr-51 as the largest contributors to the total activity while Mn-56, Co-60, Fe-55 and Fe-59 were found to be the major contributors to the dose rate. Material damage on the order of 10^{-5} DPA was estimated for the considered materials at the in-vessel irradiation position after the whole DT campaign which is sufficient to identify physical changes in the observed materials. The variation of DPA values is not significant among the considered ITER functional materials, but it has been found that the selection of the nuclear data used in the calculations can change the calculated results for up to 50% due to the differences in damage cross-sections and the threshold displacement energy values. The analysis performed within this study refers to JET experimental conditions.

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* See the author list of "Overview of the JET results in support to ITER" by X. Litaudon et al. to be published in Nuclear Fusion Special issue: overview and summary reports from the 26th Fusion Energy Conference (Kyoto, Japan, 17-22 October 2016)

O-5: Experience in the Development of Liquid Metal Plasma Facing Elements Based on Capillary Pore Structure for Steady State Operating Tokamak

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Realization of steady-state operation of fusion reactor leads to the necessity of development of essentially new design and material for plasma facing elements (PFE). In this sense the most perspective decision is the concept of use of capillary-porous system (CPS) with liquid metal that provide PFE surface self-renewal and simultaneously the closed circulation of their corrosion products. Lithium and probably its alloy with tin are considered as the most promising liquids. It has been shown that the introduction of lithium in plasma SOL provides strong screening effect for plasma pollution. At the same time, lithium is not penetrating to plasma core. It results in substantial improvement of plasma confinement and promote achievement of practically stationary modes of plasma burning. Furthermore, the problem of power exhaust with high specific density (20-30 MW/m^2) and upkeep of a comprehensible level of temperature on PFE surface can be overcame by the introduction of heat removal system into PFE design.

Experience in development, creation and experimental study of CPS based models of steady-state operating lithium PFE with systems of thermal stabilization for T-11M, T-10, FTU and KTM is considered. The possible scheme of liquid metal PFE concept realization in DEMO type and tokamak based neutron source reactors is presented.

O-6: Concept of a New Approach in Thermographic Measurements for Plasma-Wall Interaction Studies on KTM Tokamak

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Physical start-up of KTM tokamak is planned to be accomplished in the 2017. The main purpose of KTM creation is research of the first wall candidate materials for future fusion reactors under influence of plasma heat loads. Diagnostic based on methods of optical thermometry for measurements of temperature distribution on a surface of studied materials is currently being developed for KTM. Tungsten and beryllium are planned to be used as a first-wall materials in ITER. Nowadays material based on liquid lithium capillary-porous structure is actively studied as perspective plasma face component. Difficulties with measurements of metallic surface temperature in the wide range of temperature in a tokamak machine connected with: low emissivity of metals which is not known initially as a rule and can be a function of temperature, viewing direction, wavelength and physical state of the surface. Emissivity also can change due to surface modification and deposition processes, for instance beryllium deposition on the tungsten surface.

In the paper two improved complementary methods based on optical thermometry for KTM tokamak material research studies are presented. Diagnostic methods are based on using thermographic camera, two pyrometers (radiation and two colors) and CO_2 laser with 10.6 micrometer wavelength. Using these methods must improve measurement accuracy in the wide range of measured surface temperature.

In the paper described concept of non-contact temperature measuring technique is currently being developed for KTM tokamak. Plans for implementation and testing of measuring technique are also discussed.

O-7: The Superconducting Magnet System of Tore Supra: Lessons Learnt and Perspectives for WEST Steady State Operation

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Today, superconductivity permeates all large fusion projects and in particular ITER. Very low temperature components, in the range of 4 K, must be accommodated next to high temperature plasmas. A largely positive experience on such a large fusion superconducting system has been accumulated, with the satisfactory operation of Tore Supra over more than 30 years. Tore Supra introduced the use of superfluid helium at 1.8 K for a large NbTi magnet system. The fast safety discharges (FSDs) of the magnet system were the main cause of disturbance in operation. Origins of FSDs, statistics and mitigation actions are discussed.

Regarding maintenance of the system, the monitoring of the cryogenic system is a key concern and priority. A general presentation is given about the aging of the system, of the various sensors and about its overall reliability all along the operation.

A further step in this adventure is presently in progress at Cadarache with the WEST Project, consisting in transforming Tore Supra in an X-point divertor configuration with a full tungsten actively cooled divertor. The West project will take benefit of the long pulse capability offered by the superconducting magnet system, for testing the tungsten divertor. The maintenance/refurbishment phase of the cryogenic system to prepare for West operation is described. The upgraded quench detection system based on numerical technology will offer better accuracy in signal analysis.

Operating in steady state, Tore Supra played a major role to confirm the potential of fusion by magnetic confinement. It is now demonstrated that the recycled power corresponding to the refrigerator, in fusion reactors, is very moderate. The impact of plasma operations such as plasma initiation and long plasma discharges on cryogenics, is presented, highlighting very weak influence. Bath cooling by superfluid helium for the TF system of fusion reactors is a very robust and simple concept. The pros and cons of such a concept in comparison with the conventional forced flow cooling have certainly to be considered.

In December 2016, the superconducting system of Tore Supra, after a 5 year shutdown, was successfully operated at nominal current, opening the way to the commissioning of the WEST project.

O-8: Formation of the Internal Transport Barrier in KSTAR

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One of important goals of tokamak experiments is the exploration of enhanced confinement regimes, and the access of the internal transport barrier (ITB) formation is dealt with an important physics issue in the most of major tokamaks [1-3]. Also, the advanced tokamak scenario with ITB is expected to lead to a continuous reactor with high fusion power density. From this point of view, the formation of the ITB in KSTAR which is designed for long pulse operation capability is very important although its heating and current drive systems are not fully equipped yet. We have therefore assumed that an early injection of the full NBI power (~ 5.5 MW) during the current ramp-up would give a chance to form an internal barrier if the plasma could stay in the L-mode. To avoid the H-mode transition, we have produced inboard limited plasmas with detaching from the both upper and lower divertors. Using this approach, an ITB formation during L-mode has been observed which shows improved core confinement. Time trace parameters indicating the plasma performance such as temperatures, the stored energy and the β_N are comparable to the H-mode in the discharge. Ion and electron temperature profiles show the barrier clearly in the temperature, and it was sustained for about 7 s in the dedicated experiment. This is the first stationary ITB observed in the superconducting tokamak. This operation scenario with the ITB could be an alternative way to achieve a high performance regime in KSTAR, and the length of the ITB discharge could be extended even longer. In this work, we present the formation of the ITB using measured and simulated characteristic profiles.

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O-9: Transport Modeling of the DIII-D High βp Scenario and Extrapolations to ITER Steady State Operation

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Transport modeling of a proposed ITER steady-state scenario based on DIII-D high poloidal-beta (β_p) discharges finds that ITB formation can occur with either sufficient rotation or a negative central shear q-profile. The noninductive high β_p scenario has many of properties required for the ITER steady-state operation goal of a fusion gain of Q=5. The high β_p scenario is characterized by a large bootstrap current fraction (~80%) which reduces the demands on the external current drive, and a large radius internal transport barrier which is associated with excellent normalized confinement. Typical temperature and density profiles from the non-inductive high β_p scenario on DIII-D are scaled according to 0D modeling predictions of the requirements for achieving a Q=5 steady-state fusion gain in ITER with "day one" heating and current drive capabilities. Then, TGLF turbulence modeling is carried out under systematic variations of the toroidal rotation and the core q-profile. A high bootstrap fraction, high β_p scenario is found to be near an ITB formation threshold, and either strong negative central magnetic shear or rotation in a high bootstrap fraction are found to successfully provide the turbulence suppression required to achieve Q=5. Work in progress will examine self-consistent evolution of the current and kinetic profiles to test whether the strong negative central shear can be maintained.

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O-10: Exotic Plasma Shape on HL-2M

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HL-2M tokamak is now under construction in China as a modification to the HL-2A facility. A powerful plasma shape design tool has been developed on HL-2M. With this tool, not only the conventional shapes (limiter, standard single null, standard double null, snow flake and tripod divertor), but many exotic plasma shapes have been designed. Those exotic shapes include crescent, negative triangularity, leaf, and flatfish.

This shape design tool has a friendly user interface; user can easily get their target shape just by drawing or editing the plasma shape (optionally divertor leg is part of plasma shape if there is one). Up to 5000 points on the plasma shape or divertor legs are used to build the overdetermined system which has a unique minimum-norm-residual solution; this solution is the equilibrium PF current for this plasma shape. GAQ plasma current distribution model are used in this tool. In order to make sure the solution is engineering feasible; a special constraint to minimize the PF current value in the overdetermined system is applied, so the actual solution is the compromise between the target shape and the engineering limit.

The PF current and voltage waveform to reach these exotic shapes have been calculated with a plasma resistive model to estimate the resistive flux consumption. The transition between different plasma configurations (the limiter and the exotic shape) can be simply realized by linearly interpolating the equilibrium component.

The experimental demonstration of these plasma shapes will provide a valuable test bed, not only for the study of plasma confinement and transport, but also for developing control algorithm and improving control performance for fusion reactor in the future.

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IAEA
International Atomic Energy Agency

O-11: Material, Radio-Frequency and Mechanical Characterisations of High Current Steady-State Sliding Contacts for the ITER ICRH Antenna

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The ITER Ion Cyclotron Resonance Heating (ICRH) system is designed to couple 20 MW (10 MW each from two antennas) of RF power in the 40-55 MHz frequency range during 3600 s and under various plasma conditions with Edge Localized Modes. Radio-Frequency (RF) contacts are integrated within the ITER ICRH launcher in order to ensure the RF current continuity and ease the mechanical assembly. The contacts allow the free thermal expansion of the Remote Vacuum Transmission Line coaxial conductors during RF operations (peak RF current up to 2.5 kA per line at 55 MHz in steady-state conditions in the ITER vacuum environment) or 250°C baking phases. The reliability of the ICRH launcher is directly linked to these contacts, which have to withstand 30000 cycles without any routine maintenance. For this reason, the qualification of the contact materials and design requires extensive characterisations. A material and tribological experimental campaign has been carried out to determine which materials and associated coatings are relevant for RF contacts application in ITER. Following this selection, different base material samples and coating combinations have been procured and analysed (coating quality and wear resistivity). These samples have been tested in a dedicated high temperature vacuum tribometer, in which both the electrical contact resistance and the friction coefficients have been measured in vacuum and temperature environment. In parallel, RF tests have been performed with a new prototype of Multi-Contact® LA-CUT/0,25/0 contacts made of silver-coated CuCrZr louvers. During these tests, currents between 1.2 kA and 1.3 kA have been reached a few tens of time in steady-state conditions (duration longer than 60 s) without any visible damage on the louvers. Several shots have been performed at currents between 1.4 kA to 1.6 kA and during 1200 s. The RF current has been increased on the RF contacts up to a maximum of 1.9 kA during 300 s at 62 MHz, current at which the contacts suffered irremediable damage. In addition, a test bed which performs sliding test cycles has been built in order to reproduce the wear of the contact prototype after 30 000 sliding cycles on a 3 mm stroke at 175°C under vacuum. The silver coating of the louvers is removed after approximately a hundred cycles whilst, to the contrary, damage to the CuCrZr louvers is relatively low. The test beds developed during this work and the data collected gives confidence in the realization of optimized contacts for ITER.

O-12: Technological Considerations for the Steady State Operation of an NBI Beamline for Heating and Current drive

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Neutral beam injection (NBI) is attractive as an actuator for non-inductive current drive in a steady-state DEMO or fusion power plant (FPP) due to its excellent current drive efficiency. Using NBI on a power producing device demands a high wall-plug efficiency in excess of 60% that in turn requires the use of negative ions and improved neutralizer concepts such as the laser neutralizer. Furthermore, very-long-pulse or continuous operation of an NBI beamline requires vacuum pumps which do not need cyclic regeneration of a kind that would release gas into the beamline volume. All of the mentioned technologies are at the development stage. On top of that, a steady state DEMO or FPP has to have a high level of availability, and so do its components. This probably makes some degree of redundancy inevitable for the NBI system and calls for failsafe designs in such critical areas as water cooled components.

Moving from pulsed (via long pulse) to continuous operation, also means moving from an operational regime where fatigue due to cyclic loads as a lifetime-limiting failure mechanism will be replaced by wear, predominantly due to mechanisms such as sputtering, redeposition, and material damage due to neutron irradiation. This applies for example to the ion source that is prone to sputter erosion due to backstreaming ions or the high-reflectivity mirrors of a laser neutralizer that are in danger of degrading as a result of plasma–surface interaction and neutron damage.

Apart from these technological needs, the major remaining challenge regarding the physics of negative ion NBI is the stable long pulse or steady state operation of the ion source itself. With the current technology negative ion production relies on the conversion of hydrogen neutrals and positive ions on caesiated surfaces. While stable plasma generation in the rf source driver and a stable extracted negative ion current was demonstrated up to one hour at the ELISE test stand, the currents of co-extracted electrons increase steadily during long pulses. The detailed understanding of the underlying dynamics of Cs in the source and of possible solutions is, however, gradually improving.

In this paper we identify the technological and physical gaps and issues and give hints on possible solutions.

O-13: Quantitative Access to Neutral Beam Current Drive Experiments on ASDEX Upgrade

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Neutral beam current drive (NBCD) is a non-inductive current driver, suitable for steady state tokamak scenarios. Knowing the current profile and being able to predictively tailor it with NBI and other actuators helps avoiding instabilities and can be used to increase the intrinsic bootstrap current. Quantitatively predicting the NB driven current is an important prerequisite.

A fully non-inductive scenario was established in ASDEX Upgrade with I_{p} = 800 kA, q_{95} = 5.4, H_{98}(y,2) > 1.1 and β_{N} = 2.7. A fixed current value in the central solenoid avoided further inductive current drive from a time point in flattop. The current consists of ~45% off-axis NBCD, ~45% bootstrap current and ~10% electron cyclotron current drive, but the reconstructed current from the sum of the non-inductive contributions overestimates the measured plasma current slightly. This indicates the uncertainties of the simulations. The effect of a 3/2 neoclassical tearing mode that appeared in the discharge is investigated with respect to fast ion and NBCD redistribution.

Also discrepancies regarding off-axis NBCD were found earlier on ASDEX Upgrade and therefore dedicated experiments were made focus on this. Former studies comparing on- and off-axis NBCD of 5 MW NBI gave contradicting results. While MSE based current profile studies needed the assumption of anomalous fast ion transport [1], later studies of the fast ion profile constrained by the fast ion Dα (FIDA) spectroscopy were in agreement with neoclassical predictions [2]. The discrepancy could be solved in new experiments with a parallel MSE and FIDA measurement together with a revision of the NBI geometry, improved accuracy of Z_{eff}, a correction of the TRANSP radial electric field and improvements of the MSE diagnostic. The influences of fishbones and micro turbulences are taken into account and a time-, rho- and energy-dependent anomalous fast ion diffusion profile could be calculated [3] with an average diffusion coefficient below 0.3 m^2/s.

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O-14: High Field Side Lower Hybrid Launch for Steady State Plasma Sustainment

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One of the primary challenges for steady state tokamak operation is efficient, off axis current drive envisioned for advanced tokamak operation where the driven current is peaked near normalized radius, ρ, 0.6-0.8. High field side (HFS) launch of RF power in the lower hybrid range of frequencies (LHRF) in double null configurations represents an integrated solution for efficient off-axis current drive at ρ~0.6-0.8 and potentially mitigates PMI /coupling issues in present day experiments and future burning plasmas. For lower hybrid (LH) waves, wave penetration is a balance between wave accessibility ($n_{||} \propto \sqrt{n_e/B}$) and the condition for electron Landau damping ($n_{||} \sim \sqrt{30/T_e}$ with T_e in keV). The higher magnetic field on the HFS improves wave accessibility and allows for lower $n_{||}$ waves to access the plasma core, while waves launched from the LFS are restricted to the plasma periphery. A further benefit to lower $n_{||}$ launch is that the lower $n_{||}$ of waves absorbed at higher T_e yields a higher current drive efficiency. In addition to the improved wave penetration and current drive, the HFS offers additional benefits. In a tokamak, power and particles exhaust primarily to the low field side (LFS) scrape off layer (SOL) of the tokamak forcing RF launchers to be placed farther away from the plasma on the LFS, which reduces wave coupling and increases the probability of parasitic absorption in the SOL. In contrast, the HFS SOL is quiescent [Smick NF 53 023001 (2013)] and allows a smaller antenna – plasma gap that leads to good coupling as well as decreasing the likelihood of wave scattering from density fluctuations. In addition, lower electron densities measured in the HFS scrape off layer [LaBombard FN 44, 1047 (2004)] may help to reduce parasitic losses of LHRF power due to parametric decay instability. Reduced heat and particle fluxes (including neutrons) on the HFS wall will improve long-term survivability of the antenna structures. The HFS SOL also has been found to strongly screen impurities [Labombard NF (submitted)] thus mitigating adverse effects of PMI on the core plasma. Utilizing advanced ray tracing and Fokker Planck simulation tools in the LHRF regimes (GENRAY+CQL3D), simulations of reactor grade and present day experiments will be presented demonstrating efficient off-axis generation and RF antennas integrated into HFS structures.

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O-15: Impurity Seeding in JT-60SA Carbon Divertor Configuration for Efficient Power Exhaust in Steady State Discharges

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The main aim of the JT-60SA project is to analyze the near-fusion plasma conditions for support of the ITER experiment on its way towards realization of energy production in DEMO. Although in the extended phase of research the JT-60SA tokamak will be equipped with a tungsten divertor, the baseline JT-60SA design assumes a full carbon environment. In our previous studies for scenario #2 (according to the JT-60 Research Plan, high auxiliary heating power, medium electron density [1]) it was concluded that with the intrinsic impurity of C and seeded impurities N, Ne, Ar and Kr it was very difficult to reduce effectively the power delivered to the plate (P_{plate}) [2]. Moreover, seeding resulted in a very high average effective charge (about 6-8) and high impurity concentrations (> 9 %). On the other hand, our experiences with the tungsten divertor configuration show that the high-Z intrinsic W impurity can serve as an additional radiation channel in the plasma core, which helps to limit P_{plate} [3]. Therefore, this work focuses again on the carbon configuration, but beside low/medium Z impurity seeding (Ne, Ar) a high-Z impurity is added (Xe) to mimic the behavior of the W impurity in the tungsten JT-60SA configuration. The integrated core-edge COREDIV code [4] is used to analyze the influence of different seeding rate combinations of Ar/Ne and Xe on plasma conditions, especially for radiative power exhaust and impurity accumulation.

The COREDIV code describes self consistently the core and the scrape off layer with the divertor regions of tokamak plasmas. The coupling between core and edge is imposed by continuity condition at separatrix of values and fluxes of temperatures and densities. In the core the 1D transport equations with semi-empirical transport coefficients for densities and temperatures are used. Transport coefficients included the anomalous and neoclassical transport and its profiles have been modified for describing the transport barrier. In the SOL, the 2D boundary layer code EPIT is used which is primarily based on Braginskii-like equations for the background plasma and on rate equations for each ionization state of each impurity species. The sputtering processes of target material are included in the model.

Preliminary results of simulations confirm our idea that the use of gas mixtures (Xe, Ar) might solve the power exhaust problem in the low density, high power steady state JT-60SA discharges.

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O-16: Continuous Wall Conditioning in a Toroidal Device without Magnetic Field Using VHF Discharge

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The radio -frequency (RF) discharges are used for conditioning of inner walls of vacuum chambers of fusion devices. They are an essential part of operation of steady-state fusion devices. In the series of experiments on Uragan-2M the Very High Frequency (VHF) discharge [1] is investigated which needs a steady magnetic field. A regime of RF wall conditioning without the magnetic field seems is also useful for fusion devices. In presence of a confining magnetic field, the slow wave which the antenna excites is substantially slowed down in plasma that facilitates its damping. Plasma without magnetic field does not slow down the electromagnetic waves. So, to achieve acceptable damping of the wave, high electron-neural collision frequency is needed. For this reason, the RF discharge may be sustained at relatively high neutral gas pressure.

The wall conditioning is achieved due to interaction of chemically active atoms generated in the discharge with the impurities accumulated at the wall surface. The generation rate of atoms is proportional to the product of neutral gas pressure and plasma density. For the same rate of generation of atoms, at higher neutral gas pressure, the plasma density should be lower. Low plasma density is a positive factor since the probability of ionization of impurities desorbed from the wall is lower.

The radio-frequency wall conditioning without magnetic field was performed in continuous regime at the VHF frequency 130 MHz and launched power ~3kW. The discharge existed in high gas pressure 0.1...0.01Torr, was located near the antenna, and did not spread around the torus. Its parameters were measured using the Langmuir probe and the optical diagnostics. The effect of wall conditioning was judged by the amount of substances captured by the cryogenic vacuum trap. This amount appeared by the order of magnitude higher than without the discharge what indicates apparently the wall conditioning. Hydrogen, nitrogen and their mixtures had been tried as working gases. The wall conditioning in the mixture 50% / 50% was selected as the best.

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O-17: ECH Experiments and Stray Radiation Study in Wendelstein 7-X in the Context of Steady State Operation of Fusion Devices

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Electron cyclotron resonance heating (ECRH) is the main plasma heating mechanism in Wendelstein 7-X. In the first experimental campaign, up to 4.5 MW of ECRH power was available at 140 GHz, in the coming experimental campaign the ECRH power will be doubled. The unabsorbed stray radiation is distributed in the vacuum vessel.

During the first experimental campaign ECRH proved its capabilities of heating the plasma in X2 and O2, as well as driving the current. An overview of the ECRH experiments is presented.

The distribution of stray radiation in the stellarator is studied under various experimental conditions.

The distribution of stray radiation during the ECRH start-up, O2 and X2 heating, and electron cyclotron current drive experiments is considered in this contribution. During the O2 heating phase, the single-pass absorption of microwaves is better than 70%, multi-pass scheme ensures more than 90% of absorption, which is important for high density continuous operation of the stellarator. The influence of wall conditioning on the stray radiation energy flux is also discussed in this contribution.

Sniffer probes were used as an interlock in the experimental campaign of Wendelstein 7-X. The reliability of sniffer probes as a safety monitor is demonstrated.