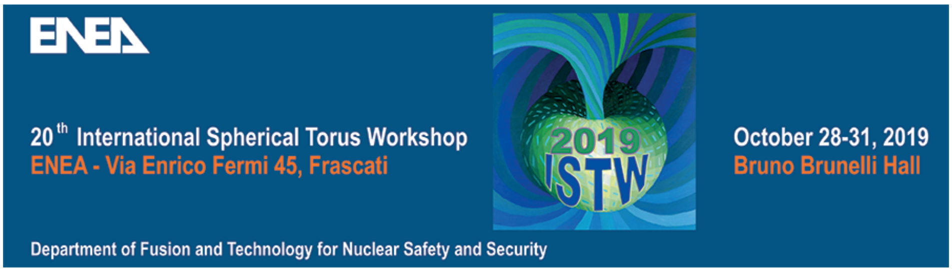
20th International Spherical Torus Workshop (ISTW2019)



Report of Contributions

https://agenda.enea.it/e/172

Contribution ID: **9** Type: **oral**

Overview of recent progress on the QUEST experiments

QUEST is a middle sized spherical tokamak in Japan and has been promoting on non-inductive plasma start-up using electron cyclotron heating (ECH) [1, 2], transient coaxial helicity injection (CHI) [3] and long duration operation of tokamak plasma using a hot wall [4]. A bifurcation of electron energy of dominantly driving plasma current was observed and the bifurcation was triggered by the reduction of fuel neutral in the fully non-inductive plasma current start-up without fundamental EC resonance. The plasma current of more than 90 kA and the density up to 5 x 1018 m-3 could be obtained with proper polarization, focusing and injection angle of RF. The EC launch from high field side (HFS) is promoting [5] and higher density than plasma cut-off could be obtained in only HFS launch. The effective mode-conversion to electron Bernstein wave (EBW) being free from density limit is expected, indicating leak RF power monitoring. More than 40kA of toroidal current was generated by transient CHI and plasma development could be predicted by a model in force-free basis [6]. Progress on fuel particle balance in the long duration discharges can be explained by a combination between hydrogen barrier model and rate equation of hydrogen (H) state [7]. The model is required to secure several parameters such as H flux to the plasma facing material (PFM) flux and recombination rate on the surface of the PFM. The technique to measure them is newly developed and has been applied in the model calculation [8]. The water cooling of the hot wall was firstly operated and was working well to modify the wall temperature and H recycling, although significant unexpected outgas of argon and nitrogen took place.

References

[1] Idei, H., et al. Nucl. Fusion 57 (2017) 126045. [2] Takase, Y., et al., Nucl. Fusion, 57 (2017) 102005. [3] Kuroda, K., et al., Plasma Fusion Res. 12 (2017) 1202020. [4] Hanada, K., et al., Nucl. Fusion 57 (2017) 126061. [5] Hatem, A., Plasma Fusion Res. (2019) to be published., [6] Kuroda, K., et al., Plasma Phys. Contr. Fusion 60 (2018) 115001. [7] Hanada, K., et al., Nucl. Mater. Energy (2019) to be published., [8] K. Hanada, et al., Nucl. Fusion (2019) to be published.

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Presenter: HANADA, kazuaki

Contribution ID: **14** Type: **oral**

Faster Fusion: ST40, engineering, commissioning, first results

Tuesday, 29 October 2019 09:35 (35 minutes)

ST40, engineering, commissioning, first results

Primary author: GRYAZNEVICH, Mikhail (Tokamak Energy Ltd)

Presenter: GRYAZNEVICH, Mikhail (Tokamak Energy Ltd)

Session Classification: Session 03

Contribution ID: **15** Type: **poster**

Status and Plans for the NSTX-U Recovery Project

The eight major scope items in the NSTX-U Recovery Project include: (1) six redesigned inner PF coils, (2) redesigned upper and lower polar region structures, (3) redesigned select plasma facing components, (4) improved bake-out, (5) additional component stress/strain trending instrumentation, (6) enhanced test cell shielding, (7) implementation of the accelerator safety order, and (8) reassembly of NSTX-U components with improved alignment. Since then, the design maturity of the Recovery scope has increased such that by this meeting >90% of the scope will have completed a preliminary design review and >70% a final design review. In addition, a comprehensive analysis of the TF bundle has been conducted to ensure that it meets the project requirements. The design, cost and schedule will be reviewed this summer and presented. Progress, status, and plans for the NSTX-U Recovery Project will be described.

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Presenter: Dr KAYE, Stanley (Princeton Plasma Physics Laboratory)

Session Classification: Poster session

Contribution ID: **16** Type: **oral**

3D HHFW full wave simulations with realistic antenna geometry and SOL plasma in NSTX-U

Wednesday, 30 October 2019 10:10 (35 minutes)

In this paper we report an application of the Petra-M on NSTX-U. Petra-M, is a recent developed and open source code, which is based on the scalable MFEM C++ finite element library and allows for FEM analysis from geometry/mesh generation, FEM assembly, FEM system equation solution, and visualization in one platform [1, 2]. The first full torus 3D high harmonic fast wave (HHFW) simulations for NSTX-U plasmas including the scrape-off-layer (SOL) region with realistic antenna geometry and core plasma will be presented. A scan of the antenna phasing is performed showing a strong interaction between FWs and the SOL plasma for lower antenna phasing, which is consistent with previous NSTX HHFW observations. The antenna spectrum for different antenna phasing will be also shown. A first attempt to couple the 3D RF solver with the full-orbit following particle code SPIRAL [3] will be discussed with the aim to show the impact of the effect of the 3D wave field on the fast ion population from NBI beams in NSTX-U.

Work supported by U.S. DOE Contracts DE-SC00108090 and DE-AC02-09CH11466

[1] S. Shiraiwa et al., EPJ Web of Conferences 157 (2017) 03048.
[2] S. Shiraiwa et al., Nucl. Fusion, in preparation (2019).
[3] G. J. Kramer et al., Plasma Phys. Control. Fusion 55 (2013) 025013.

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Presenter: BERTELLI, Nicola

Session Classification: Session 04

Contribution ID: **17** Type: **oral**

Overview of recent progress on non-inductive start-up experiment in LATE

Tuesday, 29 October 2019 10:10 (35 minutes)

LATE (Low Aspect ratio Torus Experiment) is a small device with the toroidal magnetic field up to 0.16 T at R = 0.25 m. It has no center solenoid and the plasma current is initiated and maintained by ECH/ECCD alone. There are three launchers for 2.45 GHz microwave and one launcher for 5 GHz microwave, each of which is installed on the radial port and injects microwave in the left-handed circular polarization to excite electron Bernstein wave (EBW) via O-X-B process. Overdence ST plasmas with 6 ~ 7 times the plasma cutoff density and plasma current of ~ 10 kA are formed when the fundamental ECR layer is located near the plasma core and EBW is excited in the 1st frequency band.

When EBW at the 2nd frequency band of 5 GHz is excited in the ST plasma produced by EBW at the 1st frequency band of 2.45 GHz, plasma current is driven strongly while the bulk electron parameters such as density are nearly the same. It is suggested that the injected EBW at the 2nd frequency band is absorbed mainly by high energy tail electrons in the low field side at Doppler shifted ECR and drives the plasma current while EBW at the 1st frequency band heats the bulk electrons.

Intermittent events of plasma ejection through LCFS occur in the highly overdense plasma produced by 2.45 GHz microwave. The central density decreases about 20 % and strong magnetic activity appears during the events of ~ 100 μsec. Multiple magnetic probe signals show that a current channel escapes to the upper wall and TAE like oscillations are excited. Heavy ion beam probe measurement shows that space potential near the plasma core increases about 50V during the ejection event and recovers at once.

- This work is supported by JSPS KAKENHI Grant Numbers 18H03689, 18H01198 and 17K06992.

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Presenter: Prof. TANAKA, Hitoshi (Kyoto University)

Session Classification: Session 03

Contribution ID: **18** Type: **oral**

Plasma Current Start-up and Ramp-up Experiments on the TST-2 Spherical Tokamak

Plasma current (Ip) start-up and ramp-up are being studied on the TST-2 spherical tokamak (R = 0.36 m, a = 0.23 m, Bt = 0.3 T, Ip = 0.1 MA). Both inductive and non-inductive methods are used. Up to 400 kW of RF power at 200 MHz, in the lower hybrid (LH) frequency range, is available. An innovative antenna called the capacitively-coupled combline (CCC) antenna was developed to excite a traveling LH wave with a sharp, highly directional wavenumber spectrum with the electric field polarized in the toroidal direction. Two CCC antennas are installed in TST-2, a 13-element outboard-launch antenna and a 6-element top-launch antenna. The latter was installed to improve LHW accessibility to the core and to achieve strong single-pass damping, thus avoiding the suspected wave power loss in the scrape-off layer plasma. The outboard-launch LH wave was found to be more reliable in creating the initial ST plasma, but the top-launch LH wave was found to be more efficient in subsequent Ip ramp-up. Bottom launch can be simulated using the top-launch antenna by reversing the direction of the toroidal magnetic field Bt. It was found that the bottom-launch LH wave was as efficient as the top-launch LH wave, contrary to naïve expectation, but this result can be explained by more elaborate numerical simulation.
Low voltage Ohmic Ip start-up with electron cyclotron (EC) wave assistance was also investigated. Application of EC power in the conventional Ip start-up scenario with poloidal field null extended the low pressure limit for breakdown as well as the high pressure limit for burn-through. Application of vertical field during breakdown was beneficial at low pre-fill pressure and high EC power.
AC Ohmic coil operation is a pre-ionization method in which an inductive electric field down to 0.5 V/m with a typical frequency of 1 kHz is applied. Due to the AC nature, the flux swing amplitude is about 2 orders of magnitude smaller compared to a typical Ohmic discharge. It is demonstrated that this method can be used to generate a target plasma for Ip ramp-up using either outboard-launch or top-launch LH wave.

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Presenter: TAKASE, Yuichi (The University of Tokyo)

Contribution ID: **19** Type: **oral**

Modeling of 2nd Harmonic Electron Cyclotron Heating and Current Drive Solenoid-free Start-up Experiment in QUEST

The QUEST ECH solenoid-free start-up experiment utilizing the 28 GHz gyrotron at 2nd harmonic frequency has demonstrated remarkable efficiency and achieved record start-up current values [1]. The experiment provides rich opportunities to understand and optimize ECH-based tokamak/ST current start-up and ramp-up concept. Another potentially noteworthy aspect of the QUEST 28 GHz experiment is its very high frequency to toroidal magnetic field ratio, which is 28 GHz/0.25T or 112GHz/1T. The higher frequency enables higher density limit and for reactors with several Tesla toroidal field, this start-up scenario can largely avoid the usual density limit often encountered by ECCD. Conversely this higher harmonic scenario would enable utilization of ECH at lower magnetic field as in the case of many ST experiments. This scenario maybe also attractive for the ECH assisted start up for the initial phase of ITER where the toroidal magnetic field maybe relatively low ~ 2 T. To better understand the QUEST experimental results, we initiated a modeling effort at PPPL. Improved modeling should also help develop better predictive capability for future ST and tokamak-based reactors. An ST/tokamak start-up modeling is a highly coupled non-linear problem as the magnetic field topology evolves dramatically from an open vacuum field configuration to a closed configuration. The plasma temperature evolves from a very cold collisional regime to a very hot collision-less regime. For this task, we developed a grid-based start-up code where plasma parameters, generated plasma currents, and resulting poloidal magnetic fields are evolved from the vacuum fields. Initially, 2nd harmonic electron cyclotron heating takes place with multi-pass ECH absorption as the single-pass absorption is relatively small at low temperature. The current generated in this stage is purely pressure driven since the launched wave phase and polarization information is likely lost quickly. The grad-B drift driven current together with the processional currents can then create a closed flux surface configuration and then the bootstrap current in a closed configuration can further enhance the plasma current. The ECH heating efficiency increases with plasma current since the confinement is increased and resulting electron temperature rise would further increase the ECH absorption and plasma currents. Once the plasma temperature becomes sufficiently high ~ 1 keV, a single-pass absorption can rise sufficiently to transition to the ECCD phase. The entire start-up process is therefore a self-amplifying non-linear problem where a very rapid spontaneous plasma current rise (e.g., "current jump") can be expected. An important point to note is that two-component distribution (hot minority and colder bulk) is highly advantageous for hot electrons to be generated for efficient ECCD as observed in the QUEST start-up experiment. The analysis shows that the QUEST experiment was able to generate energetic electrons by heating small hot component ~ 3% to minimize the collisional drag. Once heated to ~ 10 keV, the hot component could be sustained even with the subsequent density rise to 3-4 x 1012cm-3.
*This work supported by DoE Contract No. DE-AC02-09CH11466 [1] H. Idei, et al., IEAE 2018.

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Contribution ID: **20** Type: **oral**

Neutral particle - electron collision effect for branching plasma current ramp-up on 28 GHz 2nd harmonics ECCD

In order to realize nuclear fusion power plant, spherical tokamak (ST) which can obtain high beta has an economical advantage and is a good candidate. However, one of the problems of STs is that they not suitable for using the use center solenoid (CS) coil for the inductive current drive, because STs do not have enough cross-section in CS coil. To overcome this issue, in QUEST, the electron cyclotron resonance heating (ECRH) by the 2nd harmonic wave of 28GHz has been carried out for plasma current start-up. In 2nd harmonic ECRH, the electron energy branching from the bulk electron heating (T_e~100 eV, n_e~3×10^{18} m^{-3}, without hard X-ray radiation (HXR)) to the tail electron heating (T_e~10 eV, n_e~3×10^{18} m^{-3}, HXR energy~50 keV and high counts) was observed. In the bulk electron heating phase, plasma current is relatively low (~15 kA). On the other side, in the tail electron heating phase, plasma current was relatively high (~ 50 kA) accompanied by an increment of HXR counts. The increment of HXR counts shows the increase of tail electron. We consider that this heated electron energy branching might be caused by the shift of deposited RF energy from bulk electrons to tail electrons. The increase of high plasma current with tail electrons was observed after a decrease in H alpha. The effect of a neutral particle was confirmed by calculating the collision time of electron-electron, ion and neutral particles. By calculation, it is clear that the bulk electrons are not affected by neutral particle collision, because the effect of electron-electron, ion collision is much larger and the tail electrons are affected strongly by neutral particle collision. In addition to each collision time, the confinement time included the effect of an increase of connection length by the plasma current increase to model for each electron energy. This model showed correspondencewith the increase of confinement time by reducing neutral particles and the increase of plasma current.

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Presenter: KOJIMA, Shinichiro (Kyushu university)

Contribution ID: **21** Type: **oral**

Recent research activities on the SUNIST spherical tokamak

Recent research activities of the SUNIST spherical tokamak focused on MHD activities during ramp-up and ramp-down phase. Besides the TAEs and IREs during the ram-down phase, the tearing modes during the ramp-up of plasma current were studied in these two years. The tearing mode instabilities was carefully measured with the magnetic diagnostic system, which includes a 12-channel toroidal probe array, a 14-channel poloidal probe array at the low field side, and a 30-channel radial probe array with high spatial resolution. The mode structure appears to be typically decreasing from $m/n = -6/-1$ to $m/n = -3/-1$, where m and n are the poloidal and toroidal mode numbers. A phase reversal layer in the spatial distribution of the magnetic perturbation extracts the approximate position of the magnetic island chain, which gives insight on the evolution of the internal structure of the MHD activity. The tearing modes caused a significant loss of plasma confinement. Statistical results of the MHD response suggest that the deterioration of the plasma confinement is closely related to the phase inversion position of the poloidal magnetic perturbation.

A new fast magnetic valve has been developed for the gas injection research on the SUNIST. The gas puffing in the high field side showed a higher fueling efficiency. Also, with this fast valve, the gas outflow become supersonic molecular beam injection, which can increase the electron density more rapidly than the usual gas puffing method. The influences of fast valve gas injection in the high field side on the MHD activities and runaway current were also observed in the experiments.

Electron Bernstein wave (EBW) emission on SUNIST spherical tokamak was measured on the SUNIST. The emissions was in the frequency of 3 to 6 GHz, corresponding to the fundamental and second harmonic in low toroidal field (0.1 T) discharge. With emission intensities of different frequencies, an electron temperature profile was obtained. These measurements was also used to evaluate spontaneous conditions for EBW heating on the SUNIST□ which indicated that the spontaneous edge density profile in SUNIST is suitable for O-X-B heating scheme, and needs to be steeper for the X-B heating scheme.

The diagnostics development with high spatial and temporal resolutions on the SUNIST in recent years will also be introduced in this talk. The proposal and design of a new device□ SUNIST-2 will be brief mentioned but presented in another talk.

* This work was supported by NSFC under Grant Nos. 11827810 and 11875177 and National Ten Thousand Talent Program.

Primary author: Prof. GAO, Zhe (Tsinghua University)

Presenter: Prof. GAO, Zhe (Tsinghua University)

Contribution ID: **22** Type: **oral**

The progress of the SUNIST-2 spherical tokamak

SUNIST-2 is the successor of the SUNIST spherical tokamak (ST). The major objective of SUNIST-2 is to investigate multiple startup and heating methods of ST with higher magnetic field. The major/minor radius of SUNIST-2, of which toroidal field can be up to 1 tesla, are 0.525/0.325 meters respectively. The vacuum vessel is relatively large when compared to the size of plasmas, it has an outer diameter of 2.27 m. The extra space is reserved for the antennas of radio frequency waves. All coils of SUNIST-2 are made by copper. The toroidal field coil has 24 turns with current up to 110 kA. The central solenoid, although it is not very suitable for reactors, is kept and strengthened in SUNIST-2 as a simple but powerful startup, heating and current drive actuator. Six separately powered sub solenoids make up the central solenoid, which bring a lot of flexibilities, lower down the voltage and ease the requirements of insulation. A pair of divertor coils are attached radially at the top and the bottom of the central solenoid. The separated central solenoids and the divertor coils enable a normal ohmic discharge mode (OH mode) and a doublet mode (DB mode) for SUNIST-2. There are 6 pair of poloidal field coils in SUNIST-2. One pair of them are located inside the vacuum vessel to initiate plasmas by merging-compression (MC mode). This pair of coils can be moved vertically and keep away from the major plasma volume if other operation modes are utilized. The other five pairs of poloidal field coils are located at the outside of the vacuum vessel. The concept design and the engineering design of SUNIST-2 have been finished. The vacuum vessel and the coils are being manufactured. The power supply of coils based on super capacitors (TF) and electrolytic capacitors (CS and PF) are being tested. Basic diagnostics are being designed. The first plasma of SUNIST-2 is expected at the end of 2020.

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Presenter: Prof. TAN, Yi (Tsinghua University)

Contribution ID: **23** Type: **poster**

Measurements of electron Bernstein wave emission on SUNIST spherical tokamak

The electron Bernstein wave (EBW) is a promising method for heating and current drive in over-dense plasmas. To evaluate spontaneous conditions for EBW heating on SUNIST spherical tokamak, EBW emissions in the frequency of 3 to 6 GHz, corresponding to the fundamental and second harmonic in low toroidal field (0.1 T) discharge, were measured and analyzed. Emissions were collected by an in-vacuum quad-ridged horn antenna after mode conversion processes (B-X-O & B-X) and multi-reflections in the transmission tube, and then measured by two radiometers outside the vacuum. The edge electron density profile was measured by a Langmuir probe array and mode conversion efficiencies (TBXO & CBX) were thus calculated. The depolarization effect due to multi-reflections were evaluated through a HFSS simulation. X/O ratios measured by two radiometers were consistent with prediction values from the calculation and simulation. Emission intensities of different frequencies were calculated and then compared, and an electron temperature profile was obtained. Without absolute calibration for the antenna and radiometers, the temperature profile is relative. Results also indicated that the spontaneous edge density profile in SUNIST is suitable for O-X-B heating scheme, and needs to be steeper for the X-B heating scheme.

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Presenter: Mr WANG, SHOUZHI (TSINGHUA UNIVERSITY)

Session Classification: Poster session

Contribution ID: **24** Type: **poster**

The Current Drive Mechanism of Helicity Injection

Helicity injection is a non-inductive current drive method used in the startup of Spherical Torus and Spheromak. The mechanism of current drive is explained by Taylor relaxation by which the plasma will relax into the Taylor state. The most important feature of Taylor state is that $\lambda \equiv j \cdot B/|B^2|$ is constant all over the plasma. However, the research of how the relaxation occurs is still under progress. We propose a mechanism of Taylor relaxation through nonlinear interaction of tearing mode. We find that the nonlinear interaction of tearing mode will drive different amounts of current at different sides of rational surface, which modifies λ to be flat near the rational surface. Therefore, when many tearing modes are destabilized and overlap each other, λ will be constant all over the overlap area, which means the equilibrium state will relax into Taylor state. We also observed Taylor relaxation process in NIMROD simulation of the free decay phase of cylindrical plasma. The nonlinear modification effect of tearing modes is identified in the simulation.

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Session Classification: Poster session

Contribution ID: **25** Type: **poster**

Development of a fast-opening repetitive magnetic valve for gas injection in spherical tokamaks

Experiments have been performed on the MAST spherical tokamak that the gas puffing in the high-field side (HFS) will improve the plasma confinement. Considering the limited space and short discharge duration of SUNIST spherical tokamak, a fast and compact valve is necessary for gas injection. A new fast magnetic valve has been developed for the gas injection research on SUNIST. The open state of the valve is shorter than 1ms by the IGBT switching device used in the power supply. Besides, the valve is capable of repetitive operation during one shot of plasma discharge at the frequency of over 100 Hz. The piston made of aluminum in the valve is activated by eddy currents, and the diameter and height of which are 70 mm and 30 mm respectively, so the valve can be reliably operated on the high field side for inboard gas puffing. The flow speed of the gas can be increased to 2000 m/s in the de Laval nozzle installed on the gas outlet and the gas outflow become supersonic molecular beam injection (SMBI). The throughput which is 0.1 Pa m^3 typically for hydrogen, is adjustable by changing the pressure in the back chamber, the pulse length, and the voltage of capacitors. The fast valve has been installed and operates successfully in SUNIST. In comparison with gas puffing in the low-field side, experiments have shown that the fueling efficiency is higher in the HFS. And the SMBI can increase the electron density more rapidly than the usual gas puffing method. The influences of fast valve gas injection on the MHD activities and runaway current are also observed in the experiments.

Primary authors: Mr WANG, Binbin (Tsinghua University); Prof. TAN, Yi (Tsinghua University); Mr WANG, Zhuo (Tsinghua University); Prof. GAO, Zhe (Tsinghua University); Prof. WANG, Wenhao (Tsinghua University)

Presenter: Mr WANG, Binbin (Tsinghua University)

Session Classification: Poster session

Contribution ID: **26** Type: **poster**

Comparison of turbulent transport regulated by shear layers on conventional and spherical tokamaks

Recent flux-driven gyrokinetic computational modeling and theory suggested the existence of an E×B staircase in the plasma core consisting of a series of m/n=0/0 E×B shear layers formed at different plasma minor radii. These layers then act as semi-permeable transport barriers and thus regulate the background gradients. Experimental work in TORE-SUPRA shows that the turbulent radial correlation length exhibits a number of marked reductions across the minor radius.
In order to find more direct indications of the presence of E×B staircase, a series of experiments have been conducted on HL-2A tokamak and several solid evidences have been found. The results of turbulence analysis show several reductions of the radial correlation length and multiple shear layers of the poloidal velocity across the minor radius. The direction of poloidal velocity even reverses at different minor radii. There are also some corrugations in the ∇T_e and ∇n_e profiles at the positions of the shear layers. In this experiment, multiple plasma behaviors were studied together to give a thorough view of the E×B staircase for the first time.
However, there is rare result on spherical tokamaks about turbulent transport regulated by shear layers, especially in core plasma. Therefore, a comparison between conventional and spherical tokamak is helpful to understand the effect of poloidal asymmetry and aspect ratio on turbulent transport.
SUNIST spherical tokamak is available for turbulence research in both boundary and core plasma. The ultrafast reciprocating probe, which is used for measuring the floating potential, electron temperature, density and so on, has achieved the diagnostics in core region with slight damage on probe and weak disturbance on plasma. The density information is also obtained from the FMCW reflectometer and interferometer. The DBS reflectometer gives the poloidal velocity, which is of great importance for the research on flow shear. Additionally, the biasing electrode is used for artificially changing the flow shear.
In this work, the behaviors of turbulence transport are investigated on HL-2A conventional tokamak and SUNIST spherical tokamak and a comparison of the results is shown.

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Presenter: Mr LIU, Wenbin (Tsinghua University)

Session Classification: Poster session

Contribution ID: **27** Type: **oral**

Configuration Studies for a Low-Aspect-Ratio Liquid-Metal-Wall Sustained High-Power-Density Tokamak Facility

Wednesday, 30 October 2019 11:55 (35 minutes)

Utilizing high-Z solid walls such as tungsten is challenging for next-step/Fusion Nuclear Science Facility (FNSF)/Pilot Plant applications due to a range of issues. These issues include Plasma-Facing Component (PFC) material damage from erosion and re-deposition and neutrons, high-Z impurity accumulation and associated core plasma radiative collapse, relatively low heat flux limits at the PFC, and thermal plasma pedestal energy confinement reduction. Liquid metal walls and divertors are increasingly being studied as a possible means of addressing these challenges. However, the impact of liquid metal systems on device configuration and core plasma performance at the Proof-of-Performance level (or higher) in a tokamak configuration has not yet been systematically investigated. In this work we explore possible configurations for a sustained high-power-density tokamak facility with lower aspect ratio (A = 1.8-2.5) dedicated to the study and development of a range of liquid metal divertor and first-wall concepts. Such a device would build upon past and expected results from liquid metal test-stands, the Lithium Tokamak Experiment (LTX), the National Spherical Torus Experiment Upgrade (NSTX-U), and the Experimental Advanced Superconducting Tokamak (EAST), but the device configuration is driven primarily by the needs (space, plumbing, thermal insulation, etc.) of liquid metal systems. Configuration studies build upon previous low-A High Temperature Superconductor (HTS) tokamak pilot plant studies that incorporated a liquid metal divertor for high-heat-flux mitigation and as a means of reducing poloidal field coil current and simplifying the magnet layout and maintenance scheme. Initial physics scenario and engineering configuration studies for a next-step liquid-metal wall and divertor toroidal confinement facility are described.

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Session Classification: Session 04

Contribution ID: **28** Type: **oral**

Machine learning approaches to predict the ideal stability properties of spherical tokamak plasmas

Wednesday, 30 October 2019 11:15 (20 minutes)

Machine Learning approaches to predict the ideal stability properties of spherical tokamak plasmas

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One of the biggest challenges to achieve the goal of producing fusion energy in tokamak devices is avoidance, or mitigation, of disruptions of the plasma current due to instabilities. In order to analyse these disruptions, the Disruption Event Characterization and Forecasting (DECAF) framework has been developed [1], integrating physics models of several causal events that can lead to a disruption. Two different machine learning approaches are proposed to improve the ideal magnetohydrodynamic (MHD) no-wall limit component of the full kinetic stability model included in DECAF.
First, a powerful and partially interpretable machine learning algorithm, the Random Forest Regressor [2], was adopted to reproduce the DCON [3] computed change in plasma potential energy without wall effects, $\delta W^{n=1}_{no-wall}$. When trained on a large database of equilibria from the National Spherical Torus Experiment (NSTX), the Random Forest can significantly improve the prediction performance as well as the classification of stable/unstable points. Furthermore, this tree-based method provides an analysis of the contribution of each input feature, showing that the plasma parameters that most affect the estimated value of $\delta W^{n=1}_{no-wall}$ are the ones expected by the underlying physics. Secondly, a multilayer perceptron neural network has been trained on sets of calculations with the DCON code, to get an improved closed form equation of the no-wall limit as a function of the relevant plasma parameters provided by the Random Forest. Although being slightly worse than the Random Forest in classification performance, this approach can directly provide an estimated value of $\beta^{n=1}_{N,no-wall}$, which is key to determine the mode growth rate inside the overall kinetic stability model [4]. The model has been incorporated into DECAF and tested against a set of experimentally stable and unstable discharges. The portability of the model is also investigated, showing initial encouraging results by testing the NSTX-trained algorithm on the Mega Ampere Spherical Tokamak (MAST).

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Session Classification: Session 04

Contribution ID: **29** Type: **oral**

Validation of gyrokinetic simulations in NSTX including comparisons with a synthetic diagnostic for high-k scattering.

Wednesday, 30 October 2019 09:35 (35 minutes)

A new extensive validation study performed for a modest beta NSTX NBI-heated H-mode predicts that electron thermal transport can be entirely explained by short-wavelength electron-scale turbulence fluctuations driven by the electron temperature gradient mode (ETG), both in conditions of strong and weak ETG turbulence drive. For the first time, local, nonlinear gyrokinetic simulations carried out with the GYRO code [Candy JPP 2003] reproduce the experimental levels of electron thermal transport while simultaneously matching the frequency spectrum of electron-scale turbulence, the shape of the wavenumber spectrum and the ratio of fluctuation levels between strongly driven and weakly driven ETG turbulence conditions. Ion thermal transport is shown to be very close to neoclassical levels predicted by NEO [Belli PPCF 2008], consistent with stable ion-scale turbulence predicted by GYRO. Comparisons between high-k fluctuation measurements [Smith RSI 2008] and simulations are enabled via a novel synthetic high-k diagnostic developed for GYRO. The frequency spectra characteristics of electron-scale turbulence (spectral peak and width) can be reproduced by the synthetic spectra, but prove not to be critical constraints on the simulations. However, the shape of the high-k wavenumber spectrum and the fluctuation level ratio between the strong and weak ETG conditions can also be simultaneously matched by electron-scale simulations within sensitivity scans about the experimental profile values, and prove to be great discriminators of the simulations analyzed. Electron-scale simulations were also able to isolate the effect of safety factor and magnetic shear to match the shape of the measured fluctuation wavenumber spectrum. This work is the strongest experimental evidence to date that ETG-driven turbulence can dominate in the outer-core of modest beta NSTX H-modes.

This work has been supported by US DOE contracts DE-AC02-09CH11466 and DE-FG02-91ER54109. Computer simulations were carried out at NERSC, supported by the Office of Science of the U.S. DOE under Contract No. DE-AC02-05CH11231, and at the MIT-PSFC partition of the Engaging cluster at the MGHPCC facility (www.mghpcc.org), which was funded by DOE grant number DE-FG02-91-ER54109.

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Session Classification: Session 04

Contribution ID: **30** Type: **oral**

First operation of the Lithium Tokamak Experiment - β

Monday, 28 October 2019 12:10 (35 minutes)

LTX-β, the upgrade to the Lithium Tokamak Experiment, has now been operated with full lithium coatings on all plasma-facing surfaces, and at increased toroidal fields >0.3 T. Plasma current has so far been limited to <100 kA. The upgrade includes a neutral beam injector provided by Tri-Alpha Energy Technologies. The beam is designed to operate at 20 kV, with 35A of ion current. So far >600 kW of beam power has been injected at 18.5 kV. Up to 60% of the injected power is deposited in the plasma at higher target densities, in agreement with NUBEAM modeling. However, initial modeling of the ion orbits indicates that fast ion losses are important. Significant beam fueling is observed under some conditions. New insertable lithium evaporators have been installed on LTX-β, which provide full coverage of the plasma-facing surfaces, with a rapid (10-15 minute) evaporation cycle. Additional improvements to the evaporators, including a larger lithium inventory, and between-shots operation, are planned. LTX-β retains the same plasma geometry, and the heated high-Z liner featured in LTX. Upgrades to the diagnostic set include active CHERs. Further upgrades to the Thomson scattering system and new Lyman- α arrays will permit a determination of energy confinement time as a function of recycling, which is the emphasis for the 2019 − 2020 experimental campaign.

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Session Classification: Session O1

Contribution ID: **31** Type: **poster**

Characterization of tearing modes in NSTX using ultra-soft X-ray measurement and integrated simulations

Tearing mode (TM) instabilities are analyzed experimentally in NSTX plasmas. 2-D ultra-soft X-ray measurements are used to infer the magnetic island width, phase and radial location, by fitting the simulated perturbed emissivity to the measured one. Fitted results are further complemented by the data, on mode frequency and number from Mirnov coils and on mode frequency and location from plasma rotation. The TM parameters are then input to interpretive TRANSP simulations to test two physics models. The first model has been used for DIII-D to assess energetic particle transport caused by the magnetic islands (Bardoczi et al., Plasma Phys. Control. Fusion 2019), therefore in this work validity of the model in the low aspect ratio geometry will be tested. The second model includes an analytic representation of the island to assess its effect in general (Poli et al., Nucl. Fusion 2018). Both models, when validated, will enable predictive studies of the role of TM instabilities in integrated simulations.

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Session Classification: Poster session

Contribution ID: **32** Type: **oral**

MAST Upgrade - features and status

Monday, 28 October 2019 11:50 (20 minutes)

Presentation on the features and status of MAST Upgrade

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Session Classification: Session O1

Contribution ID: **33** Type: **oral**

Final results of the Phase-1 of the PROTO-SPHERA experiment and first evidences of plasma tori production and sustainment

The PROTO-SPHERA experiment is an unconventional magnetic confinement scheme, which aims at producing

Primary author: Dr MICOZZI, Paolo (ENEA)

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Contribution ID: **34** Type: **oral**

Final results of the Phase-1 of the PROTO-SPHERA experiment and first evidences of plasma tori production and sustainment

Monday, 28 October 2019 09:15 (35 minutes)

The PROTO-SPHERA experiment is an unconventional magnetic confinement scheme,
which aims at producing a Spherical Torus (with Ie≤300 kA) around a Plasma Centerpost (with Ie=70 kA) fed by electrodes of annular shape, in contrast with the metal centerpost of conventional Tokamaks. The Phase-1 of PROTO-SPHERA, aimed of obtaining a stable Hydrogen Plasma Centerpost, lasting 1 sec at Ie=8.5 kA longitudinal current level, was successfully ultimate at the begin of 2018. The line averaged plasma electron density on the equatorial plane of the Plasma Centerpost increases linearly with Ie and reaches, at Ie=10 kA in Argon, the considerable value of <ne>=4•1020 m-3, similar to what is obtainable today only in high-field Tokamaks, while in Hydrogen the still respectable value is <ne>=1.5•1020 m-3 at Ie=10 kA; spectroscopic measurements show very pure plasma when we operate in Hydrogen (Balmer lines only), and Langmuir probe measurement show an edge temperature of 4-8 eV and an edge density of few 1019 m-3 in proximity of the anodic plasma when we operate in Argon. The Plasma Centerpost show a strong toroidal rotation, due to E∧B, that prevents any anode anchoring. During the spring/summer 2018 the kink destabilization of the PROTOSPHERA Plasma Centerpost and the plasma tori production has already been preliminarily obtained in the actual Phase-1 of the experiment, by adding 4 impromptu vertical field external PF coils, that are wound outside the vacuum vessel and fed in series with a new power supply based on SuperCapacitors. The result in Hydrogen has been a disruption-free coupled configuration, obtained with stationary poloidal magnetic field, in which a very high aspect ratio (A=R/a≈7) and very high elongation (κ≈3) torus of approximately IST≈7 kA is sustained by Helicity Injection, with MHD bursts with a periodicity of ≈3 ms, around the Ie=10 kA Plasma Centerpost until the DC voltage VPC between electrodes is turned on, i.e. for more than 250 ms. In 2019 the PROTO-SPHERA experiment has been largely modified: a new insulating/transparent vacuum chamber in PMMA substitute the START Aluminum vessel and 6 internal PF coils has been inserted inside the machine. A "Phase-1.5" campaign is now in progress with the aim to produce/sustain, still at the Ie=10 kA level of Plasma Centerpost current, an elongated (κ≈2) low aspect ratio torus (A≤2) with a current of the order of IST≈20 kA and a volume of the closed magnetic surfaces greater than 50%.

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Session Classification: Session O1

Contribution ID: **35** Type: **oral**

Tokamak Energy and the high-field spherical tokamak route to fusion power

Tuesday, 29 October 2019 09:00 (35 minutes)

Tokamak Energy is a privately funded company based in the UK with a mission to deliver a faster route to fusion. Founded in 2009, Tokamak Energy is developing compact fusion power plants based on two promising technologies: Spherical Tokamaks (STs) and magnets made from High Temperature Superconductors (HTS). The inherent compactness and improved efficiency of the spherical tokamak, coupled with the favourable properties of HTS magnets, open a route to efficient power production at significantly lower net power outputs than previously considered possible.

Currently, two main development streams are progressing in parallel: i) advancing high field spherical tokamak physics and engineering on ST40, the highest field (BT=3T) device of its kind; and ii) HTS technology development, which has characterised tape performance and developed key technologies and is now focused on demonstrating a high field tokamak magnet system at substantial scale.

In the next stage of development, these new technologies and understanding will be combined to deliver the world's first device capable of fusion energy gain and industrial scale power production – ST-F1. ST-F1 is currently in the concept design stage and is aiming to demonstrate the viability of commercial fusion.

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Session Classification: Session 03

Contribution ID: **36** Type: **oral**

Present and future plans of PROTO-SPHERA

Monday, 28 October 2019 09:50 (20 minutes)

PROTO-SPHERA is an axi-symmetric magnetic confinement experiment, just like a spherical Tokamak: toroidal and poloidal magnetic fields (i.e. a plasma toroidal current IST) are required inside a spherical torus to contain the plasma. PROTO-SPHERA produces both fields while removing the two central metal conductors: the central toroidal rod and the central ohmic transformer; they are replaced by a plasma centerpost discharge, with helical magnetic force lines, carrying an electrode current Ie, fed by DC electrodes inside the vacuum vessel. PROTO-SPHERA however is not a Spheromak, as no metal flux conserver is present around the spherical plasma. While the presence of the toroidal field in PROTO-SPHERA is obvious, provided that the electrodes produce a stable current carrying centerpost plasma, the sustainment of the current carrying torus by DC electrodes is not obvious and it has been the first achievement of the experiment. In its initial Phase-1 (2014-2018) PROTO-SPHERA was built with 8 magnetic PF coils inside the vacuum vessel, to shape the plasma centerpost only; the electrode current reached in Hydrogen its Phase-1 limit: Ie=10 kA. By adding 4 PF coils outside the vacuum vessel, the first "thin" confined tori (that fill just 7% of the total plasma volume), endowed with divertors emerging from magnetic X-points, were formed and sustained, with Ie=10 kA and IST=7kA. The IST=7 kA toroidal current could be maintained in quasi-steady state as long as the anode-to-cathode DC voltage was applied (1 sec): it is highly plausible that the magnetic reconnections, observed around the X-points, were able to provide the steady-state current drive to the high density (>1020 m-3) confined tori. In the 2019 intermediate Phase-1.5, with Ie still limited to 10 kA, the confined tori will be compressed to low aspect-ratio, and their toroidal plasma current will be increased to IST=20-40 kA: 6 provisional internal PF compression coils have been added, bringing the total number of PF coils (internal and external) to 18. In Phase-2 (2021) the centerpost plasma current Ie will grow from 10 kA to 70 kA; the spherical shaping should allow for a confined toroidal current of IST=300 kA. Hopefully Phase-2 of PROTO-SPHERA will add to the magnetic reconnection current drive, already obtained on "thin" tori, a vigorous and self-organized plasma reconnection heating: a total power of ~20 MW will be input into the overall plasma, while the spherical tori will fill up to 95% of the total plasma volume.

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Session Classification: Session O1

Track Classification: ISTW2019

Contribution ID: **37** Type: **oral**

MAST-U experimental programme: From first plasma to first campaign

Monday, 28 October 2019 11:15 (35 minutes)

The first stage in the experimental programme of MAST-U is planned for late 2019. A modelling framework, using time-dependent vacuum field calculations, has identified viable direct induction scenarios on MAST-U that achieve suitable null quality for rapid breakdown combined with the requirements for plasma equilibrium and passive vertical stability during the early plasma current ramp up [1]. The resulting discharge parameters provide breakdown with a loop voltage of 4.5V, which around half of the total available. The loop voltage gives an electric field of 1.5-2 V/m and a null is formed with a connection length of between 400 and 800 m [1]. The null is formed with the poloidal D coils carrying currents of 5 kA or less, which is well within the capabilities of the supplies. Key to the success of MAST-U is the ability to form advanced divertor configurations, specifically the Super-X divertor [2]. Using a free boundary equilibrium solver (FIESTA) and representative current profiles, a 700 kA plasma current, double null MAST-U discharge has been simulated (κ = 2, li=0.94). The simulated discharge allows understanding of the coils required to form and control a conventional divertor and then shape this into a Super-X configuration. The elongation and internal inductance operating space has been investigated by generating a database of scenarios that will guide scenario development. MAST-U has an experimental campaign covering pedestal physics, fast particles, scenario development and exhaust physics. The focus is on the exhaust physics area, specifically understanding the advantages and disadvantages of the Super-X divertor. The increased divertor radius is thought to encourage detachment as the target temperature is proportional to 1/Rtgt2 [3]. SOLPS modelling [4] shows a reduction of the detachment threshold by a factor 2.4 by changing the target radius by 0.7m. Initial experiments will aim to test model predictions through development of conventional and super-X scenarios; utilising the extensive range of diagnostics available on MAST-U to measure the radiated power, target heat loads and temperatures.

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Session Classification: Session O1

Contribution ID: **38** Type: **oral**

Analytical studies of PROTO-SPHERA equilibria

Monday, 28 October 2019 10:30 (20 minutes)

Plasma tori formed around a 10 kA centerpost screw-pinch plasma discharge have been observed in phase-1 of PROTO-SPHERA experiments.
A simple model of plasma equilibrium has been developed, which reproduces the morphological features observed so far, in particular the bumpy shape of the centerpost discharge and the slim torus surrounding it.
The model is based on a generalization of the bumpy pinch equilibrium solution of the Grad-Shafranov equation by Taylor [Jensen T.H. and Chu M.S. 1980 J. Plasma Phys. 25 459], which allows reproducing both bumpy pinch and pinch-torus force-free configurations. The Taylor model in cylindrical coordinates is expressed as: $\psi = rJ_1(Kr) + crJ_1(\sqrt{K^2 - k_z^2}\,r)\cos(k_z z)$, where ψ is the poloidal flux function and K is the coefficient of proportionality between poloidal current and poloidal flux. The constant c determines the bumpiness of the pinch: bulgy pinches result for $c < 0$, while $c > 0$ gives pinch-torus combinations with squeezed pinch.
The above expression has been generalized along three lines. First, in order to describe low-current plasmas (including the vacuum case) it has been extended to $K < k_z$. Second, more z-dependent terms have been added, which allow obtaining bulgy pinches surrounded by slim tori. Third, a plasma pressure term $p = p_0 - \alpha\psi$ has been included. Plasma pressure in the pinch is not negligible, in fact it is sufficiently large to reverse the sign of the azimuthal current with respect to the force-free case.
Analytical solutions have also been found for the spherical G-S equation force-free conditions and with poloidal current proportional to poloidal flux (i.e. relaxed), in the form of combinations of Bessel functions of half-integer order in the spherical radial coordinate and Gegenbauer functions (for example solutions of hydrodynamical equations for symmetric Stokes flows) in the angular coordinate. These functions can be used to construct solutions of boundary-value problems for PROTO-SPHERA.

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Session Classification: Session O1

Contribution ID: **39** Type: **poster**

Implementation of the Spherical Tokamak MEDUSA-CR

The low aspect ratio spherical tokamak MEDUSA (Madison Educational Small Aspect Ratio Tokamak) built by the University of Wisconsin-Madison (USA) and donated to *Instituto Tecnológico de Costa Rica* is currently being re-commissioned at *PlasmaTEC*. The main characteristics of this device (renamed MEDUSA-CR) are: plasma major radius R_0 < 0.14 m, plasma minor radius a < 0.10 m, toroidal field at the geometric center of the vessel B_T < 0.5 T, plasma current I_p < 40 kA, $n_e(0)$ < 2.00 x 10^{20} m^{-3}, central electron temperature $T_e(0)$ < 140 eV, discharge duration is < 3 ms, top and bottom rail limiters, and D shaped plasma volume [1]. Although MEDUSA was initially constructed for educational purposes, some interesting topics may be addressed with it despite its relatively small size. It serves mainly to merge elementary synergic knowledge between the physics and the engineering involved in controlled plasma discharges and fusion related topics, which in turn could address relevant design concepts for spherical and conventional tokamaks safeguarding the cost-benefit ratio of the device operation [2]. Several topics were addressed on a first stage of engineering of MEDUSA-CR. A new vacuum system design was developed with a corresponding new design of a stainless-steel vacuum vessel instead of the original glass chamber. In addition, a new injection system was entirely developed and tested accomplishing MEDUSA's requirements. The possibility of upgrading the electric current control to an AC mode is being considered. Finally, MHD equilibrium simulations for MEDUSA-CR have been carried out using a free-boundary solver code named Fiesta (created by Geoffrey Cunninham from CCFE). This code allows to explore among other things, the best plasma-shaping scenarios regarding the *beta* ratio by changing the elongation and triangularity parameters [3].

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Session Classification: Poster session

Contribution ID: **40** Type: **oral**

Globus-M2 operation at toroidal magnetic field strength 0.7 T

Monday, 28 October 2019 15:50 (35 minutes)

Described are first results obtained on the Globus-M2 tokamak – upgraded version of Globus-M machine in spring-summer 2019 operating campaign. Operation was performed at toroidal magnetic field strength 0.7 T and plasma current 0.3 MA. Results of the campaign may be tentatively joined into a few groups. First group is related to initial stage of the discharge. Plasma breakdown conditions were improved noticeably with regard to Globus-M breakdown conditions. Stray fields at the breakdown phase became lower due to careful manufacturing and assembly of Globus-M2 electromagnetic system. The breakdown loop voltage decreases from 7 to 5 Volts (without usage of special preionization technique). Plasma ramp-up speed with 5-7 MA/s was obtained. Second group is related to OH current plateau stage and characterized by relatively low sawtooth activity. Neutral beam auxiliary plasma heating period is the third group which demonstrated very intriguing results. NBI plasma heating and current drive became more efficient at the same NB injector parameters as in Globus-M (28 keV, 0.8 MW). Plasma electron temperature during NBI pulse rose to 800 eV simultaneously with the density rise from 3x1019m-3 to 7x1019m-3. Diamagnetically measured plasma energy content increased from 3 kJ up to 6 kJ which is twice as high as in Globus-M. Ion temperature reaches 1.2 keV at the end of NB heating pulse. H-mode confinement with H factor of about 1.2 was achieved. Neutron flux rise was recorded during D beam into D plasma injection. More than doubling of neutron flux was achieved comparatively to Globus-M. And final group of results is connected with noninductive current drive. Loop voltage drop was recorded during NB injection indicating noticeable amount of non-inductively (mainly bootstrap) driven current. For the first time in spherical tokamaks noninductively driven current was recorded during LH range electromagnetic waves launch with toroidally oriented grill. During RF pulse launched by 10 waveguide grill (2.45 GHz) with the phase delay of 1200 between waveguides and total radiated power of about 150 kW ~ 30% loop voltage drop was recorded.
Achievement of Globus-M2 operation parameter limit (1 T and 0.5 MA) is planned for the later period after a full output power from thyristor rectifier supply will be accessible.

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Session Classification: Session O2

Contribution ID: **41** Type: **oral**

Thermal energy confinement at the Globus-M spherical tokamak and first results from the Globus-M2 experiments

Monday, 28 October 2019 16:25 (35 minutes)

The presentation is devoted to the thermal energy confinement study at the compact spherical tokamaks Globus-M and Globus-M2. Experiments were performed under auxiliary heating using neutral beam injection (NBI) in plasma with lower null magnetic configuration (major radius R = 0.35 m, minor radius a = 0.21-0.22, elongation ~1.9, triangularity δ~0.35) for the ranges of plasma current and toroidal magnetic field: Ip=0.12-0.25 MA, BT=0.25-0.5 T. It have been shown that energy confinement time (τE) dependence on BT is very strong, while the τE dependence on plasma current Ip is significantly weaker than IPB98(y,2) scaling predicts: τE~I_p^(0.48±0.21) B_T^(1.28±0.12). The improvement of τE was mostly by electron heat diffusivity decrease with toroidal field rise, while the ion heat diffusivity was in line with neoclassical theory predictions.
The first NBI experiments were carried out at the Globus-M2 for the increased range of plasma current and toroidal magnetic field: Ip =0.25-0.3 MA and BT = 0.7 T. During NBI heating (D-beam, 28 keV, 0.8 MW) the plasma total stored energy measured by the diamagnetic coil increased more than twice (in comparison with the Globus-M results). Diamagnetic measurements were confirmed by the kinetic (electron and ion temperature profiles) measurements. Thermal energy confinement time was estimated by 1.5D ASTRA transport modeling while the beam absorbed power was derived using two codes: NUBEAM code and 3D fast ion tracking algorithm. The obtained τE values are higher than those predicted by IPB98(y,2) and are in good agreement with the Globus-M scaling.

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Presenter: KURSKIEV, Gleb (Ioffe institute)

Session Classification: Session O2

Contribution ID: **42** Type: **oral**

CHI Research on PEGASUS 3

Monday, 28 October 2019 14:55 (25 minutes)

The spherical tokamak (ST) may require and the advanced tokamak would considerably benefit from the elimination of the central solenoid. PEGASUS-3 is a ST non-solenoidal startup development station under design and fabrication dedicated to solving the startup problem. On PEGASUS-3, Transient and Sustained coaxial helicity injection (T- and S-CHI) will be explored, as well as possible synergies of CHI with local helicity injection and EBW heating and current drive. T-CHI has shown promising capability on the HIT-II and NSTX STs. However, in both these machines the vacuum vessel was electrically cut. For reactor applications a simpler biased electrode configuration is required. To develop this capability a single biased electrode is being tested on QUEST, where up to 45 kA of toroidal current has been generated using CHI. PEGASUS-3 will use a more advanced double biased electrode configuration with optimized injector electrodes and injector poloidal field coils that should allow the T-CHI system to generate 0.3 MA of closed flux current, the limit permitted by the equilibrium PF coils. Present design indicates that standard divertor coils will provide sufficient flux for CHI studies but may be enhanced with increased current capabilities if needed. The CHI design and the CHI research plan for PEGASUS-3 will be described.

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Session Classification: Session O2

Contribution ID: **43** Type: **poster**

T-CHI current start up by a simple configuration electrode in QUEST

Transient Coaxial Helicity Injection (T-CHI) current start-up by using a simple configuration electrode installed in QUEST has been tested on a collaborative research project between University of Washington, Princeton Plasma Physics Laboratory (PPPL) and Kyushu University [1]. T-CHI has been developed as a solenoid-free plasma start-up method in HIT-II and NSTX [2,3]. The conventional T-CHI electrode in the both devices consists of inner and outer of the vessel walls which are insulated by ceramic breaks inserted in upper and lower parts of the vessel wall. The QUEST electrode is designed to be easier to introduce T-CHI to a fusion reactor, in which a bias electrode located on the lower divertor is insulated by a sandwiched ceramic break. We have examined the T-CHI discharge in two cases in which the injector region is different. When the injector flux is formed between the bias electrode and outer vessel wall, the injector current flowing from electrode is very high but the toroidal current is comparable to the injector current. The flux evolves but its footprint becomes wide. When the injector flux is formed between the bias electrode and inner vessel wall, the injector current is substantially reduced, and the plasma evolves with a narrow footprint configuration to fill the entire height of the vessel (figure 1). In addition, the plasma current pulse duration increases by several times that in the other configuration and the outer leg of the current channel is much clearly defined. The latter result has many similarities to the discharges generated by the conventional electrode on HIT-II and NSTX.
References [1] Kuroda K et al 2018 Plasma Phys. Contr. Fusion 60 115001. [2] Raman R et al 2004 Phys. Plasmas 11 2565. [3] Raman R et al 2011 Phys. Plasmas 18 092504

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Session Classification: Poster session

Contribution ID: **44** Type: **oral**

Scaling Physics of Reconnection Heating for ST Merging Startup Experiments

Tuesday, 29 October 2019 11:45 (35 minutes)

The high-power reconnection heating of merging tokamak plasma has been developed mainly in TS-3, TS-4 and MAST experiments. This unique method is caused by the promising scaling of ion heating energy that increases with squire of reconnecting magnetic field Brec. We studied mechanisms for this scaling of reconnection (ion) heating up to 2.3keV mainly using TS-3 and TS-6 experiments and PIC simulations and found the following features:
(i) the ion heating energy is as high as ~40-50% of poloidal magnetic energy of two merging tokamak plasmas and, (ii) the ion heating energy is not affected by (guide) toroidal field Bt, in the region of Bt/ Brec >1, under the two conditions: (a) compression of the current sheet to the order of ion gyroradius and (b) isolation of the merging tokamak plasmas from coils, electrodes and walls. The sheet compression to the order of ion gyroradius was found to be a key condition to realize the fast reconnection as well as the high power ion heating consistent with the Brec2-scaling prediction. Under this condition, the ion heating energy is determined uniquely by Brec ~ Bp not by Bt in the conventional tokamak operation region: Bt/ Brec >2 (or q0>2). The merging tokamak plasmas need to be fully pinched off from the PF coils without any link with coils, electrodes or walls for the purpose of minimizing loss of the hot ions heated by the reconnection/ merging. Most of the laboratory experiments of magnetic reconnection tends to have ion and electron temperatures as low as 5-30 eV due to large energy loss through magnetic fluxes intersecting coils, fluxcores, electrodes and walls. They are sacrificing the plasma confinement for the better controllability of magnetic reconnection. The ion heating was found to occur not only in the local downstream area of reconnection but also in the global merging tokamak area, increasing ratio of the ion heating energy to the electron heating energy of reconnection around 4. This promising scaling realized ion temperature as high as 2.3keV in 2019 and is expected to realize the burning ion temperature >10keV (under electron density ne~1.5x1019 [m-3]) by increasing Brec (~poloidal magnetic field Bp) over 0.6T, leading us to the high-Brec field merging tokamak experiments: ST-40 in Tokamak Energy Inc. and TS-6 in U. Tokyo.
[1] Y. Ono et al., Nuclear Fusion 59, 076025 (7pp) , (2019)

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Presenter: ONO, Yasushi (University of Tokyo)

Session Classification: Session 03

Contribution ID: **45** Type: **poster**

First Neutral Beam Injection Experiments in Versatile Experiment Spherical Torus

The neutral beam injection (NBI) system commissioning and first beam heating experiments in Versatile Experiment Spherical Torus (VEST) have been carried out. The ion beam power of ~0.6MW was successfully extracted from the ion source through commissioning. The ion source consists of arc plasma source and multi-aperture accelerator assembly which is developed by Korea Atomic Energy Research Institute (KAERI). The KAERI accelerator system was designed to deliver a 0.6MW with beam energy of 15kV. The arc plasma source can make high power of ~60kW Arc plasmas, which is enough for beam current of ~40A in this accelerator system. By using this NB system, a hydrogen NB power of up to 0.6MW was successfully injected to the VEST plasma. Difference of the plasma parameters by NB injection such as ion temperature, rotation, and plasma density was measured by using thomson scattering system, passive emission spectroscopy and so on. From these experiment results, neutral beam injection efficiency will be discussed in comparison with those of the NUBEAM simulation.

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Presenter: Mr KIHYUN, Lee (Seoul National Univertisy)

Session Classification: Poster session

Contribution ID: **46** Type: **oral**

Investigation of ion heating/transport process of magnetic reconnection during CS-free merging plasma startup of spherical tokamak in TS-6

Tuesday, 29 October 2019 12:20 (20 minutes)

Ion heating/transport process of CS-free merging plasma startup through magnetic reconnection has been investigated in the TS-3U (TS-6) spherical tokamak using ultra-high resolution 96CH/320CH 2D ion Doppler tomography diagnostics. In addition to the previously reported high-temperature plasma startup up to ∼250eV in TS-3 and ∼1.2keV in MAST, this research focused on the detailed characteristics of high guide field reconnection ($B_t > 3B_{rec}$) motivated by MAST merging/compression experiment which demonstrates promising performance for the connection to a quasi-steady scenario and pioneers a new frontier for reconnection studies: fine structure formation by reconnection heating. By identifying the double-axis field configuration with the X-point on the midplane using in situ magnetic probe diagnostics, the detailed measurement successfully revealed that the ion temperature profile forms two types of characteristic heating structure, both around the X-point and downstream. The former is affected by the Hall effect to form a tilted heating profile, while the latter is affected by the transport process which a forms a poloidal double-ring-like structure. The achieved ion heating mostly depends on the reconnecting component of the magnetic field, and the contribution of the guide field to decrease the heating efficiency tends to be saturated in the high guide field regime. Under the influence of better toroidal confinement with higher guide field, the downstream ion heating is transported vertically, mostly by parallel heat conduction, and finally forms a poloidal ring-like hollow distribution aligned with the closed flux surface at the end of merging.

[1] H. Tanabe et.al., Phys. Rev. Lett. 115, 215004 (2015)
[2] H. Tanabe et al, Nucl. Fusion 59, 086041 (2019)

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Presenter: Dr TANABE, Hiroshi (University of Tokyo)

Session Classification: Session 03

Contribution ID: **47** Type: **poster**

Analytical solutions for the guiding centers magnetically confined in a sphere

Guiding Centre theory has been developed to describe the dynamics of charged particles subject to the Lorentz force. In the context of fusion theory, it is used in numerical simulations to find approximate trajectories of particles in an external field which can be locally approximated by a straight uniform one, but also the collective effect of an ensemble of such particles as a part of a magnetized plasma.
We propose a non-perturbative formulation of Guiding Centre theory, by which it is possible to find exact trajectories of the charged particles under the influence of magnetic fields typically found in fusion devices,and in particular in spherical ones.
In this work we start by reviewing the non-perturbative formulation of Guiding Centre theory and we deduce the solutions for toroidally symmetric magnetic fields with spherical boundary conditions. Then, with the appropriate corrections to the existing codes, the particles distribution and their confinement can be computed.

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Session Classification: Poster session

Contribution ID: **48** Type: **oral**

Microwave Heating and Current Drive of Overdense Plasmas in the Upgraded Pegasus Experiment

Monday, 28 October 2019 14:35 (20 minutes)

Identifying attractive means of initiating current without using induction from a central solenoid remains a critical challenge facing the spherical tokamak (ST) concept, and is desirable for tokamaks in general. An electron Bernstein wave (EBW) heating and current drive system is being designed for use in overdense ST plasmas on the upgraded Pegasus Toroidal Experiment. The research program on Pegasus is studying the physics of non-solenoidal plasma startup, ramp up and sustainment in the ST geometry. Recent startup research on Pegasus has focused on Local Helicity Injection (LHI). This effort will be expanded upon by an EBW heating and current drive system to provide viable means for non-inductive startup and sustainment of ST plasmas. Additionally, EBW has the potential to reduce resistive losses during LHI startup by increasing the electron temperature, simplifying injector requirements, and may also offer current profile control capabilities for transition to post-startup sustainment methods. The upgraded Pegasus Experiment will increase the toroidal field by a factor of 4, allowing for EBW experiments with a source frequency of 8 GHz, providing absorption near the fundamental electron cyclotron resonance. The flexibility of the university-scale experiment allows for the ability to explore different coupling schemes such as low-field side O-mode to X-mode to EBW coupling as well as high field injection of the slow X-mode to directly couple to the EBW.

This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences under contract DE-AC05-00OR22725.

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Presenter: Dr DIEM, S.J. (Fusion Energy Division, Oak Ridge National Laboratory)

Session Classification: Session O2

Contribution ID: **49** Type: **poster**

MHD instabilities in Versatile Experiment Spherical Torus

Recent research on MHD instabilities in VEST (Versatile Experiment Spherical Torus) has focused on the tearing mode during current ramp up phase and the so-called IRE (Internal Reconnection Event) known to occur frequently in spherical torus. The tearing mode at the current ramp-up phase, known to be related to the hollow current density profile with fast ramp-up rate, seems to be suppressed when the operating gas pressure is sufficiently low. Also, the fluctuation asymmetry is observed in the internal magnetic probe measurements as well as in external Mirnov coil signals. This asymmetry is thought to be due to the coupling between two adjacent islands such as the m=2/n=1 and m=3/n=1 modes. After the current ramp-up phase, two kinds of IREs are frequently observed in VEST. Both have similar characteristics such as a large spike in plasma current, loop voltage, Mirnov signal and H-alpha emission during the IRE. However, they show significant difference in their rotation characteristics measured by ion Doppler spectroscopy. Toroidal rotation of the first kind IRE decreases to zero and becomes locked to the wall, resulting in disruption-like evolution. On the other hand, the other kind of IRE shows significant toroidal rotation acceleration in the counter-Ip direction during the IRE. Increased rotation looks like recovering the plasma current after the IRE, indicating stabilizing effect.

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Session Classification: Poster session

Contribution ID: **50** Type: **oral**

Overview of Versatile Experiment Spherical Torus (VEST): Progress and Plans

Tuesday, 29 October 2019 11:10 (35 minutes)

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Session Classification: Session 03

Contribution ID: **51** Type: **oral**

Transient CHI start-up of low inductance plasma for advanced ST scenarios

Wednesday, 30 October 2019 11:35 (20 minutes)

Demonstration of fully solenoid-free start-up at the ~1 MA level and ramp-up to full current of low inductance plasma would be a major step for the ST program, as it would be a convincing enough step for the consideration of ST designs with a reduced size solenoid. In support of this objective, the TSC code, with the inclusion of plasma transport, has been used to study the Transient CHI closed flux current start-up potential in a BT =1 T NSTX-U device and at increased BT =3 T, to assess the potential capability on ST-40, as the ST-40 vessel dimensions are comparable to those of NSTX-U.

In reactors based on the ST/AT concepts, minimizing the recirculation power fraction is desired for an attractive and economical electrical power generation system. Such devices must operate at high fractions of the bootstrap current drive. The ST configuration with a low aspect ratio is particularly attractive in this regard. In addition, to sustain the high-performance regime, density and pressure profile control should be demonstrated using reactor relevant core fueling systems that during each fueling pulse does not significantly perturb the optimized profiles necessary for maintaining the bootstrap current drive.

The immediate advantage of solenoid-free start-up is that it provides more space to add structural material to the toroidal field coil to support a higher peak field and higher field on axis − which could greatly increase fusion performance and/or reduce the cost of the facility. Reducing the size of the central solenoid would allow the valuable in-board region to be used for other more critically important systems that are necessary for sustained steady-state operation. Some of these new capabilities are highspeed pellet injection from the inboard side (without the use of a guide tube with strong curvature) and high field side RF launchers for heating and current drive.

During transient CHI start-up, as well as during the merging-compression start-up pioneered on the START and MAST devices, all the start-up current is produced during the initial plasma formation pulse, and one does not continue to drive current after the plasma has formed. Consequently, the scaling of these concepts to larger devices is relatively well understood as a reliable predictive model is difficult to develop as the plasma response during the current ramp-up is not well known, especially at high current.

Transient CHI TSC simulations at 1 T show that closed flux start-up currents >500 kA should be achievable with the present divertor coil set on NSTX-U. NSTX-U is also well equipped with high power neutral beam injection capability that is optimized for sustained non-inductive current drive of the CHI target. Simulations also show that if the BT is increased to 3 T, and with increased divertor coil current rating, closed flux start-up currents more than 0.6 MA should be possible. Work is in progress to assess the requirements for 1 MA closed flux current start-up in a 3 T device. TSC simulations also show that at these high levels of magnetic flux injection the CHI plasma self-heats to 100s of eV electron temperature. Additional heating, if required, could be provided by ECH as the CHI plasma electron density is below the density limit for ECH systems considered for STs. The ST-40 device would be a particularly attractive device on which to demonstrate MA level current start-up followed by sustainment with reactor-relevant ECH.

Primary author: Dr RAMAN, Roger

Presenter: Dr RAMAN, Roger

Session Classification: Session 04

Contribution ID: **52** Type: **oral**

Technical Benefits of the PROTO-SPHERA Confinement Configuration

Monday, 28 October 2019 10:10 (20 minutes)

The PROTO-SPHERA experiment can produce plasma spherical tori with a simply connected configuration (the centerpost is practically implemented by the plasma itself) and a high confinement efficiency (β→1). Such configurations are obtained from the self-organization of a plasma arc (screw pinch) in a magnetostatic field. This approach leads to several technical and economic advantages, that even exceeded the initial theoretical expectations.
The experimental system can be operated by only 3 groups of power supplies (for cathode heating, poloidal field and pinch, respectively) with a virtually constant current and with a limited demand of electrical energy. In fact, the recent experiments showed that the plasma tends to a toroidal (tokamak-like) shape even without the very fast current derivatives predicted in the initial design. The plasma current is induced and confined without a toroidal magnet and without variations in the magnetic flux and in the poloidal coil currents. The maximum voltage applied in the structure, even at the arc breakdown, does not exceed 350 V. A high temperature can be reached without additional heating and current drive systems.
Axisymmetric double null configurations with electron density in the order of 1020 m-3 were already observed. The experiments also showed that the PROTO-SPHERA configurations are stable as long as the pinch voltage is applied. The plasma is not terminated nor disrupted even in presence of sudden deformations of the pinch. Therefore, the set-up is consistent without any further coils or radiofrequency sources to cope with plasma instabilities.
The vessel is cylindrical and it can be totally opened and closed in few days for maintenance and modifications. For example, the adjustment of the poloidal field by the introduction of new internal and external coils is relatively simple, as already done with several external coils during the experimental campaigns. Since 2019 the middle part of the cylindrical vessel is non-conducting and transparent in order to avoid undesired couplings for the plasma arcs or for the magnetic field produced by the external coils and in order to make the plasma configuration clearly visible for the researchers and for the diagnostic tools.
The insertion and expulsion of elements, as pinch current or gas, through the configuration follow a simple and well defined path. The fusion products could be ejected through a magnetic nozzle on the cylinder axis (with possible applications in spacecraft propulsion).

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Presenter: LAMPASI, Alessandro (ENEA)

Session Classification: Session O1

Contribution ID: **53** Type: **oral**

Integrated Studies of Solenoid Free Tokamak Startup

Monday, 28 October 2019 14:00 (35 minutes)

The Pegasus Toroidal Experiment has focused on exploring the unique features of extremely low aspect ratio operation to study plasma stability at A ~ 1 and near-unity beta, including access to ohmic H-mode with associated ELM behavior and plasmas with an absolute minimum-B topology. For such studies, the Local Helicity Injection (LHI) technique was developed to initiate and drive toroidal current without use of a central solenoid. LHI utilizes compact, edge-localized current sources (Ainj ≤ 8 cm2, Iinj ≤ 8 kA, Vinj ≤ 1.5 kV), and has initiated > 0.2 MA of plasma current at low-field (BT ~ 0.15 T) and near-unity aspect ratio (A). Recent studies have addressed comparisons of LHI injector locations, MHD characteristics, confinement properties, and the transition to ohmic sustainment. The Pegasus program is now terminating and being replaced with an upgraded facility called Pegasus-III (for Pegasus-Phase III). This new program is focused on developing an understanding and comparative studies of several leading candidates for non-solenoidal startup techniques. These include: DC helicity injection, including LHI and sustained and transient coaxial helicity injection (S-CHI and T-CHI); electron Bernstein wave (EBW) radiofrequency electron heating and current drive; and poloidal field induction. Future capabilities may include electron cyclotron heating and current drive and/or neutral beam current drive. Upgrades include: a new solenoid-free centerstack assembly to provide a 4x increase in BTF to 0.6 T; a new high-stress TF coil system; increased pulse length to study confinement and current evolution; expanded power systems; enhanced diagnostics; and an expanded lab facility. Plasma startup will be enabled by: advanced LHI injectors with shaped electrodes and active helicity injection control; a refractory metal electrode system supporting S-CHI and T-CHI without vacuum electrical breaks or current flow through the vacuum vessel; and an expanded poloidal field coil set. An 8 GHz, ~400 kW EBW system will be added. A goal of the new experiment is to demonstrate non-solenoidal startup to p = 0.3 MA and provide guidance for MA-class startup applicable to NSTX-U and ultimately a nuclear facility. This program includes collaborations from the University of Wisconsin, University of Washington, ORNL, and PPPL.
*This work is supported by U.S. Department of Energy grants DE-FG02-96ER54375 and DE-SC0019008.

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Presenter: FONCK, Raymond (University of Wisconsin-Madison)

Session Classification: Session O2

Contribution ID: **54** Type: **poster**

Enhanced Pedestal H-mode Regime on NSTX

The largest normalized energy confinement attained over a confinement time on NSTX (H98 > 1.5) was achieved in the ELM-free Enhanced Pedestal (EP) H-mode regime [1-3] that features a wide pedestal with a significant increase in the edge carbon temperature and rotation gradients. These discharges achieve improved energy and momentum confinement with a beneficial decrease in the impurity accumulation relative to a standard ELM-free H-mode regime on NSTX [3]. Access to EP H-mode is improved with reducing wall recycling via solid lithium wall coatings and by operating at large Ip. EP H-mode was often observed following the recovery from a large ELM and terminated with another ELM or core MHD activity.
Linear CGYRO [5] calculations indicate the particle and electron energy transport are predominately due to TEMs in the steep gradient region of the pedestal, while previous GS2 calculations [6] for wide pedestal discharges imply ETG modes contribute to the electron energy transport at the bottom of the pedestal. The ion energy transport throughout the pedestal is in good agreement with the neoclassical transport exceeding the anomalous transport. EP H-mode is often triggered by a transient period of lower edge ion collisionality where ÑTi increases due to reduced neoclassical ion thermal transport. The reduction in ion energy transport leads to an increase in the anomalous transport in all channels as the Te profile is stiff, consistent with linear stability calculations, BES measurements [7] and reduced 1-D transport models [4]. The increase in anomalous particle transport, combined with a reduction of the impurity pinch at lower collisionality, reinforces the lower edge density and can drive a positive feedback if the improvement in the neoclassical energy confinement at low collisionality exceeds the degradation from the increased anomalous transport.
The favorable positive feedback initiated by transiently accessing lower ion collisionality in the wide pedestal regime motivates the development of actuators for controlling the edge density that are compatible with large core density and heat flux mitigation on NSTX-U.
*Supported by the US DOE under DE-AC02-09CH11466 & DE-FC02-08ER54977.
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Session Classification: Poster session

Contribution ID: **55** Type: **oral**

NSTX-U: Recent Results and Plans

Wednesday, 30 October 2019 09:00 (35 minutes)

NSTX-U: Recent Results and Plans
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After an 18-month sequence of key reviews, the NSTX-U Recovery Project was recently approved to initiate procurement and fabrication of components as a step towards completion of the Recovery and commencement of physics operations in the 2021-2022 time frame. The elements of the Recovery, which include new PF coils, new support structure including center stack casing, redesign of graphite tiles with improved heat flux handling capabilities, will insure highly reliable operations. The first few years of operation will focus on two primary goals: assessing ST performance at up to 6 times lower collisionality than could be achieved in NSTX, and development of fully non-inductively sustained plasmas for pulse lengths up to 5 s. The longer-term goal is the development of a full toroidal deployment of liquid lithium divertor modules to assess this transformative solution to handling the high heat fluxes expected in next-step devices. During the Recovery outage, research has targeted topics from NSTX/NSTX-U data or through collaborations on other devices that could facilitate the achievement of the NSTX-U research goals once operation starts. These topics include understanding mechanisms that control pedestal, core and fast ion transport and stability, and developing associated predictive models, developing real-time control capabilities and wall conditioning. This presentation will describe the recent progress in this research.

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Proposed Liquid Metal PFC Development Program in NSTX-U

Projected divertor heat flux footprints from Eich1 and Goldston2 scalings for next step devices is in the range of 1-2 mm, with unmitigated peak heat fluxes of several hundred MW/m2. However there may be meaningful deviations from these scalings for spherical tokamaks, e.g. NSTX3. At full plasma current and heating power, nonetheless, H-mode discharges in NSTX-U4 are projected to have midplane heat flux widths of 2-3mm, with resulting unmitigated peak heat fluxes of 30 MW/m2. Coupled with the scientific need to confirm and understand these scalings, especially the possible role of turbulence to increase SOL heat flux footprints5, the NSTX-U divertor physics program will both measure the scalings at full parameters, as well as embark on a liquid lithium PFC development program both as a power exhaust strategy and a method to increase energy confinement toward compact reactor designs.
The long-term goal of this liquid Li PFC program is to deploy full toroidal coverage flowing liquid Li PFCs by the latter part of the 2020's, using full high-Z Mo-based (TZM) tiles as substrates. Li coatings and other technologies would be used to cover/coat the Mo tiles outside of the divertor. Experience with static liquid Li PFC technology at modest inventory levels would be obtained by implementation of pre-filled Li tiles, which were designed for NSTX-U and tested in MAGNUM-PSI6, 7. Experience with flowing liquid Li PFCs would be obtained by installation of a flowing liquid Li midplane limiter, building on three generations of collaborative design and experimentation on the EAST device8. This proposed plan would mesh seamlessly with the nascent US liquid metal PFC development program aimed at developing concepts for a fusion nuclear science facility or a compact pilot plant. *Work supported by U.S. DoE contract DE-AC02-09CH11466 with PPPL.
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