Relation of pedestal stability regime to the behavior of ELM heat flux footprints in NSTX-U and DIII-D

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It is found that pedestal plasma regime plays an important role in the behavior of ELM heat flux footprints. The effective area at the divertor surface that accommodates total power expelled by ELMs, the wetted area (A_{wet}), is crucial for the determination of ELM mitigation requirements for ITER. A_{wet} has been observed to broaden significantly during the ELM, compared to the inter-ELM values, in several tokamaks [1] where the pedestal plasma is usually against the peeling-ballooning stability boundary with intermediate to high toroidal mode number (n=10–20). However, in NSTX [2] and DIII-D, cases of significant contraction of A_{wet} by ELMs have been observed when the pedestal plasma is against the current driven kink/peeling boundary with low toroidal mode number (n=1–5). Data in DIII-D have shown that increasing pedestal collisionality (ν_e*=0.3 → 0.9 → 3.5) leads to more favorable ELM footprints behavior, i.e. from contraction toward broadening, being consistent with an ELITE analysis that shows movement of the operating point in the stability diagram from the peeling toward the ballooning side, i.e. most unstable for n=5 → 10 → 25, respectively. NSTX pedestal plasma is found to be dominantly on the peeling side [3], with ELMs usually reducing A_{wet}, although the pedestal ν_e* is generally higher than ~1 contrary to DIII-D. It is suspected that strong shaping in the ST geometry may have played an important role in determining pedestal stability in NSTX. Non-linear ELM simulation using BOUT++ is in progress to compare to the observed footprints behavior. ELM data from NSTX-U will be presented and compared to the DIII-D results. This work was supported by the U.S. DOE, contract numbers DE-AC05-00OR22725 (ORNL), DE-AC02-09CH11466 (PPPL), DE-AC52-07NA27344 (LLNL), and DE-FC02-04ER54698 (GA).

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First results of plasma central column in PROTO-SPHERA

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The PROTO-SPHERA experiment aims at replacing the metal center post of a Spherical Tokamak with a plasma central column. The plasma of the machine can be considered to be composed by three distict regions: an upper "mushroom-shaped" anode plasma, shaped by 4 constant current poloidal field (PF) coils internal to the cylindrical vacuum vessel; a lower symmetric cathode plasma; the intermediate plasma central column. The first phase of PROTO-SPHERA, in which 8 constant current shaping PF coils have been built along with their power supply, with the cathode heating and with the central column plasma power supplies, is producing (break-down voltage is 75 V in Ar, 200 V in H) the central column plasma only, with a current of 1.7 kA (8.5 kA are due): the current is limited by spurious plasma discharge paths near the vacuum vessel wall, which are driving half of the plasma current on the outboard of the main path, this problem is being cured by adding 4 further constant current PF coils, external to the vessel. The first results of PROTO-SPHERA have already removed the major concern that the central column plasma could attach itself on a restricted portion of the annular anode (both electrodes were designed with annular shape in order to handle the high plasma power and current density impinging upon them). Each PF coil inside the vacuum vessel is in principle at a floating electric potential, but can as well be connected through resistances either to the anode or to the cathode or to the vessel (ground) potential. Each PF coil is spontaneously and indipendently charged to an electric potential by the plasma discharge itself: luckily enough the ensuing electric field **E** inside the machine produces an **E∧B** drift which distributes smoothly the plasma on the annular hollow anode, with neither evidence of attachment nor of filamentation in the anode plasma region. This result is even more impressive as the plasma emerges from the directly heated annular cathode in 18 filaments, as the cathode in the first phase of PROTO-SPHERA is only partially filled with 18 groups (instead of 108) of 3 superposed Tungsten emitters, but the **E∧B** drift eliminates the filamentation just as the plasma enters the anode region.

Study of edge turbulence and L-H transitions in NSTX Ohmic plasmas

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An important goal in fusion research, of late, is to develop a predictive understanding of the transition from L-mode to H-mode. In the new paradigm of drift waveï zonal flow turbulence [1-2], it has been emphasized that a significant portion of the available free energy from the gradients (∇n and ∇T) can be deposited in the zonal flows. Such models, however, need rigorous experimental validation for efficient operation of future fusion devices like ITER and beyond. Hence, the turbulence at the plasma edge needs to be analyzed in high spatial and temporal resolutions simultaneously. Characteristics of the turbulence in the edge and scrape-off-layer (SOL) of Ohmic plasmas in NSTX are studied in this work using data from the gas puff imaging diagnostic (GPI) [3]. Two dimensional turbulence velocity fields are derived from imaging velocimetry based on orthogonal dynamic programming (ODP) algorithm [4]. From the time dependent traces of the turbulence poloidal velocity (v_Z) across the L-H transition, it is apparent that the increase in v_Z precedes the fall in D_α signal and RMS fluctuation level. Hence, it is possible that the poloidal flow has initiated prior to the L-H transition. Further, a new ~40 kHz coherent mode is found in Ohmic L-mode plasmas for the first time in NSTX. Dynamics of the turbulence-zonal flow system across the L-H transition and statistical features of the blob dominated transport in L-mode plasmas will be reported.

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Bifurcation to Enhanced Pedestal H-mode on NSTX

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The bifurcation from H-mode (H_{98} < 1.2) to the ELM-free Enhanced Pedestal (EP) H-mode (H_{98} = 1.2 - 2.0) on NSTX occurs when the ion thermal (χ_i) and momentum transport become decoupled from particle transport, such that the ion temperature (T_i) and rotation pedestals increase independent of the density pedestal [1, 2, 3]. The onset of the EP H-mode transition is found to correlate with decreased pedestal collisionality (ν^*_{ped}) and an increased broadening of the density fluctuation (dn/n) spectrum in the pedestal as measured with beam emission spectroscopy. The spectrum broadening at decreased ν^*_{ped} is consistent with GEM simulations that indicate the toroidal mode number of the most unstable instability increases as ν^*_{ped} decreases [4, 5]. The shifting of the ion-scale turbulence to higher frequencies (spectrum broadening) may describe the decrease in the thermal transport versus particle transport. The lowest ν^*_{ped}, and thus largest spectrum broadening, on NSTX-U is achieved with low pedestal density via lithium wall conditioning and when Z_{eff} in the pedestal is significantly reduced via large edge rotation shear from external 3D fields or a large ELM. This is consistent with the observation that EP H-mode is often triggered by a large ELM event and most often occurred when using lithium-conditioned walls. Kinetic neoclassical transport calculations (XGC0 [6]) confirm that Z_{eff} is reduced when edge rotation braking leads to a more negative E_r that shifts the impurity density profiles inward relative to the main ion density. These calculations also describe the role kinetic neoclassical and anomalous transport effects play in the decoupling of energy, momentum and particle transport [7] at the bifurcation to EP H-mode. This work was supported in part by U.S. Department of Energy contracts DE- AC02 - 09CH11466.

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Modifications to Ideal Stability by Kinetic Effects for Disruption Avoidance

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Marginal stability points of global modes during high beta operation in NSTX can be found by computing kinetic modifications to ideal magnetohydrodynamic limits on stability. Calculations with the DCON code for nearly five thousand experimental equilibria show that the no-wall beta limit decreased with increasing aspect ratio and increasing broadness of the pressure profile, which has implications for NSTX-U. Kinetic modification to ideal limits calculations for several discharges as computed using the MISK code predict a transition from damping of the mode to growth as the time approaches the experimental time of marginal stability to the resistive wall mode. The main stabilization mechanism is through rotational resonances with the ion precession drift motion of thermal particles in the plasma, though energetic particles also contribute to stability. To determine RWM marginal stability for use in disruption avoidance, ideal stability limits need to be modified by kinetic effects in order to reproduce experimental marginal stability points. Guided by the full calculations, reduced stability models are investigated to inform automated disruption characterization and prediction analyses presently being developed using NSTX data for application to NSTX-U.

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Full Wave Simulations of Fast Wave Scrape-off Layer Losses of NSTX/ NSTX-U in Mid/High Harmonic Regime and a Comparison with C-Mod/EAST in the Minority Heating Regime

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Recent experimental studies of high harmonic fast wave (HHFW) heating on the National Spherical Torus eXperiment (NSTX) have demonstrated that substantial HHFW power loss can occur along the open field lines in the scrape-off layer (SOL) when edge densities are high enough that the fast waves can propagate close to the launcher. For several decades, RF modelling codes have neglected the presence of the SOL, concentrating instead on understanding the wave dynamics in the core plasma bounded by the last closed flux surface LCFS. In this work we examine fast wave propagation and power loss in the SOL of tokamak plasmas by using the full wave code AORSA, with the edge plasma beyond the LCFS included in the solution domain and with a collisional damping parameter used as a proxy to represent the real, and most likely nonlinear, damping processes. 2D and 3D AORSA results for the NSTX, show a strong transition to higher SOL power losses (driven by the RF field) when the FW cut-off is removed from in front of the antenna by increasing the edge density. Predictions for NSTX-Upgrade (NSTX-U) indicate similar result obtained for NSTX. However, an important difference is that the transition to high losses is predicted to occur at higher densities (about a factor two) in NSTX-U, indicating a wider SOL density range in which the experiment can run with lower SOL power losses. This result will be further verified in the upcoming NSTX-U experimental campaign. As a comparison, full wave simulations have been extended to "conventional" tokamaks with higher aspect ratios, such as Alcator C-Mod and EAST devices, which unlike NSTX/NSTX-U that operate in the mid/high harmonic regime, operate in the minority heating regime. In the minority heating regime AORSA results indicate lower SOL power losses with increasing density in front of the antenna, in agreement with the experimental observation that increasing the density in front of the antenna leads to better antenna-plasma coupling.

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ECH/EBW SYSTEMS FOR ST STARTUP AND CURRENT DRIVE

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Lower frequency (~15-30 GHz), high power microwave generation and transmission systems have been of interest for plasma initiation and non-inductive current drive during the startup phase of ST operation. Some success has been achieved with ECH/EBW startup experiments conducted on the MAST ST at Culham [1] and EBW heating experiments performed on the linear plasma Proto-MPEX [2] experiment at ORNL utilizing older 28 GHz gyrotrons. Medium power EBW systems have been proposed in the past for NSTX and more recently high power 28 GHz systems (~ 1 MW) have been proposed for NSTX-U [3]. Electron Bernstein Wave plasma modeling on these devices generally shows reasonable coupling efficiency for up to the second harmonic. Launcher beam optics analysis favor higher frequency for improved spot size. Waveguide transmission systems were developed using 88.9 mm corrugated aluminum waveguide components and associated compact launchers. Larger diameter waveguide prototypes of up to 114 mm diameter have been developed that could operate at both 28 GHz and a lower frequency such as 19 GHz depending on which frequency is of interest. Other Gyrotron hardware is available for experiments in the 28 -53 GHz range at power levels of 300-500 kw. Hardware options for various devices and test measurements and experimental results for existing EBW/ECH experiments will be discussed.

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H-mode and ELM Dynamics Studies at Near-Unity Aspect Ratio in the PEGASUS Toroidal Experiment and their Extension to PEGASUS-Upgrade

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Tokamak operation at near-unity A provides access to advanced tokamak physics at modest plasma and device parameters. Ohmic H-modes are routinely attained on the PEGASUS ST, in part due to the low L-H power threshold P_{LH} arising from low-B_T operation at $A\sim1$. Their characteristics include: improved τ_e, with $H_{98}\sim1$; edge current and pressure pedestal formation; and the occurrence of ELMs. Recent experiments have examined magnetic topology and n_e dependencies of P_{LH}. P_{LH} exceeds the ITER L-H scaling by 10–20×, with P_{LH}/P_{ITPA08} increasing sharply as $A \to 1$. No P_{LH}-minimizing n_e has been found. Unlike at high-A, P_{LH} is insensitive to limited or diverted magnetic topology to date. Modest pedestal values at $A\sim1$ afford unique edge diagnostic accessibility to study ELMs and their nonlinear dynamics. $J_{edge}(R,t)$ measured through a Type I ELM shows a complex pedestal collapse and filament ejection. These studies are being extended to higher I_p and longer pulse length with LHI non-solenoidal startup to improve MHD stability.

An upgrade to the PEGASUS ST, PEGASUS-Upgrade, is planned to exploit low-A characteristics and diagnostic accessibility to support the validation of the physics basis needed for ITER and beyond. Unique studies will be pursued in three areas: local measurements of pedestal and ELM dynamics at Alfvénic timescales; direct measurement of the local plasma response to application of 3D magnetic perturbations with high spectral flexibility; and extension of LHI startup to NSTX-U relevant confinement and stability regimes. Significant but relatively low-cost upgrades to the facility are proposed to support them: a new centerstack with larger solenoid and doubled TF conductors; a new TF and reconfigured OH power supplies; and installation of an extensive 3D magnetic perturbation coil system. PEGASUS-Upgrade will provide 0.3 MA plasmas with pulse lengths of 50–100 ms I_p flattop, aspect ratio < 1.25, and toroidal field up to 0.4 T.

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Development and Plans for Feedback Control of Stored Energy, Vertical Position, and q_0/l_i in NSTX-U

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The upgrade to the National Spherical Torus Experiment (NSTX-U) adds a larger center-stack, enabling higher toroidal field and longer pulse duration, and three new neutral beam sources aimed more tangentially than the existing three, increasing available heating and current drive, and allowing deposition profiles to be tailored. In order to exploit these new capabilities and meet the high-performance operational goals of NSTX-U, advanced feedback control algorithms will be required. Major upgrades to the Plasma Control System (PCS) have been made and several algorithms are under development to improve and add capabilities, including shape control, vertical position control, beam source modulation, stored energy control, and profile control. To facilitate studying the performance of the control schemes that are under development, a framework for conducting closed loop simulations in the integrated modeling code TRANSP has been developed. The framework exploits many of the predictive capabilities of TRANSP and provides a means for performing control calculations based on user-supplied data (controller matrices, target waveforms, etc.). Among the first planned feedback control activities is the control of the stored energy and the current profile. Based on predictive TRANSP simulations, control-oriented models for the dynamic evolution of these quantities have been developed and used for model-based control design. Simulations of a control algorithm for simultaneous control of β_N and q_0 using total beam power and the outer gap using the TRANSP framework will be presented to demonstrate the framework. Progress towards application of the approach to non-inductive scenarios and an outlook for application to the design of an ST based Fusion Nuclear Science Facility will also be discussed. Status of the NSTX-U control system and preliminary results from the first NSTX-U run campaign will be discussed.

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M3D-C[1] Simulations of the Plasma Response to Externally Applied Magnetic Perturbations in NSTX-U Snowflake Divertor Configurations

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Understanding the nature of the plasma response to small resonant and non-resonant 3D magnetic perturbations applied by non-axisymmetric coils is essential to suppress or mitigate Edge Localized Modes (ELMs) without significantly impacting the plasma performance or core stability. However, even with complete suppression or mitigation of ELMs achieved, the handling of the exhaust power can still be problematic in reactor-size machines like DEMO since conventional divertor solutions are expected to be insufficient to keep the divertor heat loads within the operating limits of the plasma-facing components [1]. The "snowflake" (SF) divertor [2] is one of several magnetic divertor configurations that have been proposed as a potential way to reduce the divertor heat loads. Therefore, it is also essential to investigate the effect of 3D magnetic perturbations in plasmas with a SF divertor configuration. This work reports on resistive magnetohydrodynamic simulations, performed using the M3D-C[1] code [3], of the plasma response to $n = 3$ magnetic perturbations applied to NSTX-U plasmas with various SF divertor configurations. The simulations are performed for SF configurations with various distances between primary and secondary x-points, various values of ion effective charge (Z_{eff}) and for various values of current in the NSTX-U non-axisymmetric magnetic perturbation coils. Preliminary results show that higher values of Z_{eff} tend to reduce the plasma response. Single- and two-fluid plasma simulations will be presented and compared with calculations based on the vacuum approach.

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Long-pulse operation of the PFRC-2 device

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Studies of the time dependence of plasma density, photon emission, power coupling, and fluctuation levels in long-duration plasma pulses were performed in the PFRC-2, a field-reversed-configuration device heated by odd-parity rotating magnetic fields. Long-pulse operation is made possible by a set of eight superconducting internal passive flux conserving rings, each with an inductive decay time of 1 sec and a critical current of 3 kA. With prefill hydrogen gas only, the line-average density rose to $2x10^{12}$ cm^{-3} in 1 ms and decayed to near 0 in about 10 ms. Using a PV-10 gas valve modified to provide supersonic gas injection, we have found operational regimes where in-discharge fueling with a single 1-ms-duration hydrogen puff produced stable high density ($2x10^{12}$ cm^{-3}), warm (T_e > 200 eV) plasma discharges that persisted for over 200 ms.

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Neutronics Analysis of HTS-ST

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Spherical tokamaks (STs) have the potential to provide a cost-effective approach to fusion power and as a neutron source for experimentation, as they hold the plasma in tighter magnetic fields; forming a more compact reactor requiring lower capital construction and material costs. However, due to the smaller physical size of the tokamak and the aspect ratio that characterises its geometry, there is a limited amount of inboard space in and around the centre column. The restricted space limits the size of the centre column and means that only a limited amount of shielding can be used to reduce the nuclear heating and damage levels experienced by superconducting coil-type ST's. Since the associated cryoplant in a power device would need to remove the power deposited in the central column, at what is expected to be a significant cost per kW, this aspect of the superconducting ST design is a key economic consideration.

The limited inboard space also restricts the tritium breeding blankets to the outboard. For a self-sustaining tritium supply, for the D-T fusion reaction, a tritium breeding ratio of >1.1 is required.

The radiation transport code MCNP (Monte Carlo N-Particle) and inventory code FISPACT have been used to perform neutronics analysis on a HTS-ST model based on the PPPL FNSF radial build and CAD model. This work draws on previous analysis, using simplified parameterised ST models, on the centre column with respect to the material composition and size of the centre column, shielding material and shielding thickness. Initial neutronics results will be presented.

Reconnection physics and fast flux closure during simulations of Coaxial Helicity Injection in NSTX/NSTX-U

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Magnetic reconnection, which energizes many processes in nature, has been shown to have a fundamental role in the plasma start up and current formation in NSTX/NSTX-U. Through resistive MHD simulations, it is demonstrated that during transient Coaxial Helicity Injection (CHI) discharges at high Lundquist number, the elongated current sheet formed through a Sweet-Parker forced reconnection process [1] breaks up, and a transition to spontaneous reconnection (plasmoid instability) occurs. [2] Consistent with theory, fundamental characteristics of the plasmoid instability are demonstrated through NIMROD MHD simulations of transient CHI experiments in the NSTX. Simulations have been performed in a realistic geometry with a toroidal guide field and using experimental NSTX poloidal coil currents. Motivated by the simulations, experimental camera images have been revisited and suggest the existence of reconnecting plasmoids in NSTX. As CHI is a promising candidate for plasma start-up and may ultimately also have the potential for steady-state current drive, it is thus important to understand the CHI physics to be able to correctly model it in simulations of NSTX/NSTX-U and to be able to extrapolate its viability to a reactor. Our simulations show that plasmoid-mediated reconnection may be the leading mechanism for fast flux closure.

A large-volume flux closure during transient CHI experiments in NSTX-U is also demonstrated through MHD simulations. Several major updates, including the location of CHI poloidal coils, are planned to improve the CHI start-up phase in NSTX-U. Simulations in the NSTX-U configuration with fixed coil currents show that with strong flux shaping the injected open field lines (injector flux) could rapidly reconnect and form a large volume of closed flux surfaces. This is achieved by driving parallel current in the injector flux coil and oppositely directed currents in the flux shaping coils to form a narrow flux footprint and push the injector flux. As the helicity and plasma are injected into the device, the oppositely directed field lines in the injector region (a) are forced to reconnect and form a current sheet (b) or spontaneously to reconnect when the elongated current sheet becomes MHD unstable. Simulations in NSTX-U also show that the magnetic pressures around the enclosed flux surface would support a steady configuration to allow a good start-up equilibrium after the injector voltage is turned off. Supported by DOE-FG02-12ER55115.

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Shielding and Breeding Considerations for ST-Based HTS-FNSF Design

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The shielding and tritium breeding assessments for the ST-based fusion Nuclear science facility (FNSF), being addressed at the University of Wisconsin-Madison through a national collaborative effort led by the Princeton Plasma Physics laboratory, represent key elements for achieving the design engineering objectives. These include adequate protection of the electrically efficient high-temperature superconducting (HTS) magnet against radiation and tritium self-sufficiency using outboard-only blanket as much as practically possible.

HTS magnets are potentially attractive for fusion applications due to the high operating temperature (20-40 k) (that reduces the cryogenic heat load), high current density, and high magnetic fields at coil (approaching 20 T). The most capable, radiation-resistant HTS option for ST applications is REBCO that could tolerate up to $3x10^{22}$ n/m^2 fast neutron fluence ($E_n > 0.1$ MeV) and allow 5 mW/cm^3 of peak nuclear heating without damaging the HTS magnet.

Numerous shielding and cooling materials have been examined to select the optimal shield that primarily protects the inboard magnet of the 3 m major radius device. The plasma generates 560 MW of fusion power and 1.1 MW/m^2 machine average neutron wall loading, producing significant neutron fluence (~6 MWy/m^2) at the outboard midplane for blanket and materials testing during the 10 year of operation. The potential impact of the candidate inboard materials (ferritic steel, tungsten carbide, hydrides, water, borated water, and heavy water) on shielding the magnet as well as reflecting neutrons to the outboard blanket to enhance the tritium breeding ratio (TBR) has also been assessed.

The blanket of choice is the dual-cooled lithium lead (DCLL) blanket – the preferred blanket concept in the US for future devices. High-pressure helium coolant and flowing LiPb remove the surface and nuclear heating and circulate to external heat exchanger and tritium extraction systems. Our 3-D neutronics model included the details of blanket internals, the long-leg divertor configuration, five outboard test modules, and several H/CD ports. An additional effort was made to examine the need for a thin inboard blanket to achieve an overall TBR in excess of unity with a wide margin.

LSP simulations of fast ions slowing down in cool magnetized plasma

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In MFE devices, rapid transport of fusion products, e.g., tritons and alpha particles, from the plasma core into the scrape-off layer (SOL) could perform the dual roles of energy and ash removal. Through these two processes in the SOL, the fast particle slowing-down time will have a major effect on both the energy balance of a fusion reactor and its neutron emissions. In small field-reversed configuration (FRC) devices, the first-orbit trajectories of most fusion products will traverse the SOL, potentially allowing those particles to deposit their energy in the SOL and eventually be exhausted along the open field lines. However, the dynamics of the fast-ion energy loss processes under conditions expected in the FRC SOL, where the Debye length is greater than the electron gyroradius, are not fully understood. What modifications to the classical slowing down rate are necessary? Will instabilities accelerate the energy loss? We use LSP, a 3D PIC code, to examine the effects of SOL plasma parameters (density, temperature and background magnetic field strength) on the slowing down time of fast ions in a cool plasma with parameters similar to those expected in the SOL of small FRC reactors.

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Suppression of TAE and GAE with HHFW heating

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This talk reports on analysis of experiments where ≈3 MW of high-harmonic fast wave (HHFW) heating was successful in completely suppressing energetic particle modes (fishbones), toroidal Alfvén eigenmodes (TAE) and global Alfvén eigenmodes (GAE) activity. Previous experiments have explored the use of RF to suppress frequency chirping, but here we see complete suppression of the modes. The suppression is not immediate, but takes some 10's of ms, suggesting that the HHFW is modifying the distribution of fast ions, or possibly other equilibrium plasma parameters. Neutron rate and NPA data show no indication of excessive fast ion losses with HHFW, although NPA data suggests some phase-space redistribution is happening. The target plasma was not usual (low plasma current Helium plasmas), future experimental work will explore extending this suppression technique to more typical conditions.

Overview of edge modeling efforts for advanced divertor configurations in NSTX-U with magnetic perturbation fields*

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Symmetry breaking effects such as resonant magnetic perturbations (RMPs) present a challenge for the numerical analysis of divertor operation, because they require three dimensional models. One such model is provided by the EMC3-EIRENE code, which is based on a finite flux tube grid for field line reconstruction that allows to account for realistic, three dimensional configurations. The family of "snowflake" divertor configurations includes a variety of magnetic topologies. We present an overview of different "snowflake" configurations and their interaction with an externally applied RMP field with base mode number n = 3 at NSTX-U. Furthermore, we present the Field Line Analysis and Reconstruction Environment (FLARE) - a collection of tools for the analysis of the magnetic field structure. It includes a flexible grid generator which allows to set up plasma transport simulations with the EMC3-EIRENE code. It also includes a set of tools to analyze the quality of field line reconstruction in the transport code, and we will present a comparison between a low aspect ratio NSTX-U configuration and a high aspect ratio DIII-D configuration.

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Plasma equilibrium and fluctuation measurements in the plasma edge using a Rogowski probe in the TST-2 spherical tokamak

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Measurement of the local current density in the plasma edge region in Ohmic plasma discharges was performed successfully using a Rogowski probe in the TST-2 spherical tokamak ($R_0 = 0.36$ m, $a = 0.23$ m, $B_t < 0.3$ T) [1]. The Rogowski probe consists of two multi-layer Rogowski coils, five magnetic field pick-up coils and two Langmuir probes, and it can be moved along the major radius of the tokamak and rotated around the shaft axis (in the toroidal-poloidal plane). Thus, the measurement of the current density profile in the major radial direction, including the angular dependence in the toroidal-poloidal plane, can be accomplished. The current density profile was measured in outboard-limited and inboard-limited Ohmic plasma discharges. In TST-2, the antenna limiter on the outboard side is located at $R_{ant\text{-}lim} = 585$ mm. In inboard-limited plasmas, current was observed behind the limiter (i.e., $R > R_{ant\text{-}lim}$), and the measured current density profile did not agree with the profile calculated by EFIT. On the other hand, for in inboard-limited plasmas, such current was not observed, and the measured current density profile agreed with the calculated profile.

Significant positive spikes and oscillations in both positive and negative directions were observed in the measured local current. Using a fast camera (up to 1,000,000 frames per second) and a Langmuir probe placed on the Rogowski probe, visible light emissions from the probe and ion saturation current spikes were observed at the time of the current spike. It is inferred that current spikes are due to a locally dense filamentary structure passing through the Rogowski coil.

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Observation of divertor peak heat flux reduction by turbulence during the inter-ELM and ELM-free phase in NSTX

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Recent research indicates that the midplane Scrape-off Layer (SOL) power fall-off length (λ_q) for ITER is expected to be very narrow, ~1mm, for the baseline inductive H-mode burning plasma scenario at I_p=15MA [1]. This result is consistent with the prediction from a heuristic drift-based theory [2]. However, turbulence effect has not been considered in the heuristic model and it is important to develop a more comprehensive understanding of physical processes that determine λ_q. In NSTX, filamentary structure in heat flux profile, suspected to be induced by turbulence blob, is observed during the inter-ELM and ELM-free phase for many H-mode discharges. The filamentary heat flux structure varies in time; beginning to deposit heat onto the near SOL and then further out to the far SOL at a later time during the cycle. In the beginning of ELM-free phase, the heat flux profile is narrow and no filamentary structure is observed. With the appearance of blob turbulence, the heat flux width increases by ~100% with total power deposited to the divertor surface almost constant. This results in the reduction of peak heat flux (q_{peak}) by >50%. Filamentary structure in heat flux profile is often observed during the inter-ELM periods and this leads to the reduction of q_{peak} by ~50% compared to that during the no-blob ELM-free phase. The temporal evolution of filamentary structure is consistent with the gas-puff imaging (GPI) diagnostic data. Detailed data analysis for the physical processes of heat deposition by turbulence blobs with a range of operating parameters from NSTX will be presented. This work was supported by the US DOE, contract number DE-SC0008309.

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Toroidal Alfven Eigenmode Driven by Runaway Electrons During Minor Disruptions in the SUNIST Ohmic Plasmas

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The toroidal Alfven eigenmode (TAE) excited by runaway electrons during minor disruptions was identified in the SUNIST Ohmic plasmas. The TAE mode was observed in the frequency range 150-400kHz, with the m/n=-3/-1 and -4/-1 harmonics, and propagating in the electron diamagnetic direction in the laboratory frame of reference. This mode appeared only when a runaway plateau was built in the post-disruption plasma, which is quite different to the MHD behavior during the internal reconnection events (IREs). The resonant interaction of runaway electrons with the precession drift frequency was considered to drive the TAE. The potential application of mitigating runaway electrons by exciting the TAE is discussed as well.

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NSTX-U Facility and Project Plans

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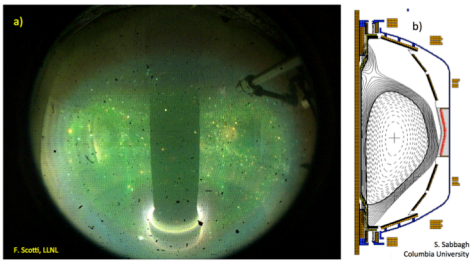
The construction of NSTX-Upgrade has concluded, and the final preparations for research operations are underway. This upgrade to the NSTX facility was motivated by the desire to study plasmas with significantly lower collisionality, very high non-inductive current fraction, and high divertor power fluxes with innovative divertor geometries, among other goals. In order to achieve these aims, the central magnet assembly has been replaced with coils sufficient to provide 1 T toroidal field (0.55 T on NSTX), plasma currents up to 2 MA, and more flexible divertor geometry. Additionally, a 2nd neutral beam system, containing three neutral beam sources and capable of injecting up to 7.8 MW of total power at 100 keV, has been installed. This neutral beam system has significantly larger tangency radius than the otherwise identical 1st beamline, providing larger neutral beam current drive for non-inductive sustainment. Significant steps in these engineering projects will be described in this talk.

The completion of the neutral beam upgrade was demonstrated by the successful operation of a neutral beam source at 45 kV on 5/11/2015. Completion of the center-stack upgrade project was demonstrated by the generation of approximately 100 kA of plasma current on 8/10/2015. Plasma currents up to 140 kA were achieved in subsequent operations; note that a full vacuum-vessel bakeout was not performed for these first plasmas. Highlights of these operations will be described, including first lessons learned about the operation of the new facility.



a) Image and b) poloidal flux plot for 120 kA plasma from NSTX-Upgrade.

The NSTX-U team is now transitioning to completing preparation for research operations. This involved a final round of coil testing, diagnostic calibrations, and vessel bakeout, culminating in an operations restart in early October. A detailed sequence of experiments has been developed to bring on the full capabilities of the facility. These commissioning steps, as well as diagnostics upgrades and first plasma results, with be described.

In the longer term, significant additions to the NSTX-U facility are envisioned. The long term facility plan calls for converting all PFCs to high-Z materials. As a first step, the tiles on the horizontal divertor targets will be replaced with TZM molybdenum after the first run campaign. In later years, a 28 GHz gyrotron is planned in order to provide electron heating during the start-up phase, furthering the goals of fully non-inductive current ramp up. A lower divertor cryo-pump is under design, to provide density control for long-pulse scenarios. Finally, a set of in-vessel off-midplane 3D field coils is being designed. This will dramatically expand the capabilities of NSTX-U for RWM control and RMP & NTV studies. The motivation and status of these facility upgrades will be described.

Merging-compression formation

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Merging-compression plasma formation is a solenoid-free plasma formation method used in STs. Two plasma rings are formed and merged via magnetic reconnection into one plasma ring that then is radially compressed to form ST configuration. Plasma currents of several hundred kA and plasma ion and electron thermal temperatures of keV-range have been produced on START and MAST tokamaks using this method, however full understanding of the physics of all three stages of m/c formation has been obtained only recently, and will be presented here in detail for the first time. Plasma rings formation and adiabatic compression can be explained and modelled using physics well-known in tokamak research and experimental data from MAST and START can be used to justify predictions of the values of achievable plasma currents in future devices. The reconnection process can be described by resistive MHD and two fluid models, using slow shock and current sheet models [D.Biscamp, Phys Lett, 105A 1984 124], however only recently these models have been benchmarked with the experimental data. All models predict strong ion and electron heating at increased toroidal (guide) and poloidal (reconnection) fields.

This method will be used to create ST plasmas in a compact (R~0.4-0.5m) high field (3T) ST40 tokamak which is under construction, expecting operations in 2016. Main parameters, physics and engineering issues will be discussed in detail as well as update on the construction status. Moderate extrapolation from MAST/START data suggests achievement of plasma current of ~2MA. Using theoretical predictions benchmarked on experimental data from TS-3, TS-4, USTX, MAST and START, 10keV range temperatures should be achieved at densities ~ 0.5-1.0x10^20 m^-3, bringing ST40 plasmas into burning regime conditions straight from the plasma formation, assuming that the energy confinement in ST has strong favorable dependence on TF.

Analysis and prediction of momentum pinch in spherical tokamaks

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Previous perturbative measurements in NSTX H-modes indicated the existence of an inward momentum pinch. Assuming that $\Pi_\varphi = nmR^2(-\chi_\varphi \nabla\Omega + V_\varphi \Omega)$, where Ω is the measured toroidal angular rotation rate, Π_φ is the momentum flux determined from TRANSP and V_φ is the pinch velocity, the pinch parameter was found to be RV_φ/χ_φ = (-1) – (-7) [1]. Local, quasi-linear gyrokinetic predictions of ITG instability including the Coriolis drift [2] provide a reasonable explanation of the momentum pinch in conventional tokamaks [3]. However, the Coriolis effect predicts near zero or outward pinch for the NSTX H-modes (in contrast to experiment) due to kinetic ballooning modes that are predicted to be unstable at relatively high beta (β_T=12-16%) [4].

To provide an additional test of momentum pinch theory at low aspect ratio, plans were made to perform similar experiments in MAST L-mode plasmas at relatively low beta, thereby eliminating the complexity of electromagnetic effects at high beta. These experiments were conducted during the final MAST campaign (2013) using applied n=3 RMP fields to perturb the plasma rotation. The time-dependent response of the rotation after the RMP field is removed is used to infer both momentum diffusivity and pinch. The inferred pinch values, RV_φ/χ_φ=(-2)-(-11), are similar to those found in conventional tokamaks and NSTX H-modes. Gyrokinetic predictions are beginning to determine if a discrepancy in predicted momentum pinch is found at these lower beta conditions. A similar L-mode experiment is planned for NSTX-U in the upcoming run. This work is supported by US DOE contract DE-AC02-09CH11466, and the RCUK Energy Programme and EURATOM.

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Recent progress on non-inductive current drive and particle balance control towards steady-state operation on QUEST

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Non-inductive plasma current start-up is a crucial issue in development of spherical tokamaks (STs) and steady state operation (SSO) is a common issue in further development of cost-effective fusion power plants. The QUEST project aims to obtain SSO of STs with non-inductive current drive and well-controlled hydrogen recycling. QUEST has a capability to inject three kinds of RFs of 2.45GHz, 50kW CW, 8.2GHz, 400kW CW and 28GHz, 500kW 1sec, to drive non-inductive plasma current and has newly equipped a hot wall (up to 673K) inside of the vacuum vessel to control hydrogen recycling. Recent progress towards SSO on OUEST will be introduced.

Non-inductive plasma start-up without fundamental electron cyclotron resonance (1^{st} ECR) was successfully demonstrated in both 28GHz [1] and 8.2GHz [2]. Especially the plasma current was finally ramped up around 50-60kA and it had good reproducibility with 28 GHz RF. A comparison of their efficiency between with and w/o 1^{st} ECR was performed with the 8.2 GHz RF system and the efficiency to drive current and produce plasma w/o 1^{st} ECR is significantly inferior as compared with 1^{st} ECR heating and soft X-ray measurements in different energy ranges shows inefficient extendibility of electron energy at the ECR layer. Numerical calculations is well-explained by a poor capability to electron acceleration at higher harmonics ECR. An experimental survey of the efficiency on toroidal field (B_T) gives us a further possibility to gain plasma current in higher B_T. It is expected to drive more than 60 kA with the 28 GHz RF in higher B_T.

Fuel hydrogen behavior in plasma-facing wall (PFW) has been investigated with measuring injecting and evacuating hydrogen molecules (H_2) in the plasma-producing vessel of QUEST. Approximately 80% of the wall-storing hydrogen atoms (H) is almost out-going within 300s after termination of the discharge, which indicates dynamic retention is dominant. A newly proposed particle balance equation to express hydrogen recycling on dynamic retention dominant wall is successfully confirmed in long duration discharges. The equation predicts to significantly control hydrogen recycling with the hot wall and the first experimental results will be presented.

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Effect of the thermal particles collisionality on resistive wall mode in tokamak

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Analytic results indicate that the kinetic damping of trapped thermal particles on resistive wall mode (RWM) significantly depends on the particles collision frequency (ν_{eff}). In this work, it is predicted that there exist two branches of the RWM when the kinetic damping of thermal particles is taken into account. Here, both trapped thermal ions and electrons are included in the model. When ν_{eff} is smaller than a critical value, plasma flow (ω_0) induced drift kinetic resonance tends to stabilize one branch of the mode but destabilize the other one. When collisionality is over the critical value, increasing of the flow velocity does not generally change the stability of either of the branches. The possible mechanism is discussed as follows. When $|\nu_{eff}| \sim |\omega_d + \omega_0 - \omega|$, either $|\nu_{eff}|$ or ω_0 can strongly affect the perturbed kinetic energy through the mode-particle resonance, which in turn influences the mode instability; Otherwise when $|\nu_{eff}| \gg |\omega_d + \omega_0 - \omega|$, $|\nu_{eff}|$ has dominant contribution to the perturbed energy, which is then insensitive to ω_0. Here, ω_d and ω denote the precession frequency of trapped particles and the mode frequency, respectively. Both the energy-independent and energy-dependent collision models yield the similar conclusion. In addition, self-consistent computations from MARS-K agree well with the analytical predictions, and show that the particles collisionality slightly modifies the mode structure. This work indicates that in the certain region of collision frequency, the stability of RWM is sensitive to the rotation frequency. Future experiments in low collisionality plasmas, such as NSTX-U and ITER, may find that the RWM stability is more sensitive to the changes in rotation than present day, high collisionality devices.

Status and Plan for Versatile Experiment Spherical Torus (VEST)

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Ohmic operation of VEST (Versatile Experiment Spherical Torus) has been successfully carried out, generating plasma currents of up to 70 kA with pulse duration of ~15 ms at the toroidal magnetic field of 0.1T on axis. The elongation and edge safety factor of the low aspect ratio plasmas (R=0.4m and a=0.3m) are estimated to be 1.6 and 3.7 respectively from equilibrium reconstruction based on magnetic diagnostics.

Various plasma start-up experiments have been conducted in VEST. Efficient ECH (Electron Cyclotron Heating) assisted plasma start-up scheme utilizing the trapped particle configuration (TPC), a mirror like magnetic field configuration, is developed. TPC significantly enhances the pre-ionization with the enhanced particle confinement due to its mirror effect, and stable decay index in this configuration also enables the prompt plasma current initiation. Consequently, TPC demonstrates robust tokamak start-up with lower loop voltage with lower volt-second consumption as well as wider operation range in terms of the ECH pre-ionization power and the H2 filling pressure. Electron gun has been installed to study electrostatic helicity injection as another startup scheme. By using single power configuration that can provide voltages for both injector operation and helicity injection, gradual rising plasma current of ~ 10kA has been achieved with injection current and voltage of 1.5kA and 500V respectively. With additional power supply for higher injection voltage, plasma currents of up to ~30kA have been achieved with the peak injection current and voltage of 1.0 kA and 1.5 kV respectively.

Various wave heating and current drive experiments are under preparation such as EBW (Electron Bernstein Wave) and low hybrid fast wave heating experiments. Direct mode conversion of X-mode to EBW from low field side is clearly observed in the pre-ionization stage and will be extended to heating and current drive. To study advance tokamak regime with high beta and high bootstrap operation in VEST, high power neutral beam and profile diagnostics are under preparation. Meanwhile ohmic plasmas will be improved to provide proper target plasmas for the high power neutral beam by increasing plasma density with higher plasma current at longer pulse length.

Toward Active Current Density Profile Control in NSTX-U: Performance Assessment via Predictive TRANSP Simulations*

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Active control of the toroidal current density profile is among those plasma-control milestones that the NSTX-U program must achieve to realize its operational goals. An overview of current efforts toward current-profile control in NSTX-U will be provided. A two-component control design approach is followed for the regulation of the current profile by actuating the total plasma current and the powers of the individual neutral beam injectors. First, nonlinear optimization techniques are used to compute offline actuator trajectories that achieve specific plasma scenarios. Given a desired operating state, the optimizer produces the actuator trajectories that steer the plasma to such state. The objective of the feedforward control design stage is to provide a systematic approach to scenario planning. Secondly, a feedback control algorithm is developed to track a desired current density profile evolution by modifying online the previously computed feedforward trajectories. The objective of the feedback control design stage is to add robustness against model uncertainties and disturbances to the overall current-profile control scheme. Both the actuator-trajectory optimizer and the real-time feedback controller are designed based on a first-principles-driven, control-oriented model that predicts the evolution of the current density profile by combining the poloidal magnetic flux diffusion equation with empirical correlations obtained at NSTX-U for the electron density, electron temperature, and non-inductive current drives. The proposed control scheme is being tested in TRANSP simulations through the recently developed Expert routine, which provides a framework to perform closed-loop predictive simulations within the TRANSP source code. Simulation results assessing the effectiveness of the proposed control scheme will be presented.

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Center-Solenoid-Free Merging Start-up of STs by Outer PF coils in UTST

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A center-solenoid-free [1] merging start-up scheme for spherical tokamak plasmas was developed in a University of Tokyo spherical tokamak (UTST) experiment by using outer poloidal field (PF) coils. Null points are formed transiently in the upper and lower sections of the UTST vacuum vessel by combination of outer PF coil currents and wall eddy currents. Torus breakdown was successfully initiated at the null points and two spherical tokamak plasmas with a total current up to 80 kA were simultaneously generated by induction from outer PF coils [2]; however, only 40 % of the vacuum magnetic flux at the null point was conserved in the ST plasma. Low flux utilization efficiency and high electric field of >40 V/m required for breakdown are the weak points to be improved and optimization of the plasma initiation process is under investigation by using numerical modeling of torus discharge at a transient null point.

Merging process of the two ST plasmas provided substantial ion and electron heating by magnetic reconnection. It was found that about 30 % of the initial poloidal magnetic energy was released during merging. Then, about 80 % of the released energy was converted to ion thermal energy and 20 % to electron thermal energy. The obtained dependence of heating on plasma current suggests that high temperature and high-current plasma suitable for neutral beam injection is attainable under the realistic conditions in the merging start-up method.

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0.8-5 keV X-ray emission from PFRC-2 plasma

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Using a Si-detector pulse-height method, we measured the X-ray spectrum emitted by odd-parity rotating-magnetic-field (RMF_o)-driven hydrogen plasma in the PFRC-2 device. At RMF_o powers below 10 kW absorbed, the spectrum appeared Maxwellian with $T_e \sim 200$ eV and $n_e \sim 10^{12}$ cm^{-3}. The X-ray count rate was largely constant over a 5 ms pulse. It peaked at an axial field near 50 G and decreased as the magnetic field increased, reaching zero at fields over 100 gauss. The count rate was also dependent on the parameters of the capacitively coupled low-density (10^{10} cm^{-3}) low power (20 -400 W) seed plasma which contained 0.1% energetic electrons with T_e up to 500 eV. Here, we present x-ray count rates and electron temperatures from the PFRC-2 plasma with different background field strengths, RMF_o powers, and seed plasmas, as well as limited axial and temporal profiles.

3D Modeling of NSTX Vertical Displacement Events with M3D-C1

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Nonlinear free-boundary 3D tokamak resistive wall instabilities are calculated using a new resistive wall model [1] in the two-fluid M3D-C1 3D magnetohydrodynamics code. In this model, the resistive wall and surrounding vacuum region are included within the computational domain. This implementation contrasts with the method typically used in fluid codes in which the resistive wall is treated as a boundary condition on the computational domain boundary using the thin-wall approximation. Besides allowing a finite thickness, and possibly a non-axisymmetric wall, this approach has the advantage of maintaining purely local coupling of mesh elements. This new capability is used to simulate the fully 3D nonlinear evolution of vertical displacement events (VDEs) in NSTX. The VDE calculations are performed in diverted tokamak geometry, at physically realistic values of dissipation, and with resistive walls of finite thickness. When the vertical control system is switched off, the plasma begins to drift, initially axisymmetrically, with a vertical growth rate proportional to the wall resistivity. After a brief period, as the plasma drifts and deforms, a toroidal mode number n=1 resistive wall mode with dominant poloidal mode number m=2 begins to appear, and it too initially grows with a growth rate proportional to the wall resistivity. Eventually, as the plasma becomes limited by the first wall, the growth rate of the n=1 mode increases dramatically leading to a rapid thermal and current quench. Strong (induced) currents are observed to flow in the wall and between the plasma and the wall (halo). New diagnostics are being implemented to facilitate comparisons between these results, previous M3D and TSC simulations, and NSTX magnetics data. These simulations have recently become more efficient as M3D-C1 now has the capability of restarting a 3D calculation from a 2D one, so the axisymmetric phase of the VDE can be done very efficiently in 2D.

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Status and plan of LHFW(Lower Hybrid Fast Wave) current drive research in VEST

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The slow wave in LH (Lower Hybrid) resonance range has been utilized as the most efficient current drive method in tokamaks. However, the density limit and strong electron Landau damping of slow wave make it hard to penetrate into the core plasma region in a reactor grade plasmas. In this respect, the fast wave branch of LH waves could be a good candidate for the central electron heating and current drive due to its penetration and damping in more high density plasmas. To investigate the feasibility of LHFW current drive, analytic study and ray tracing simulations have been carried out on VEST (Versatile Experiment Spherical Torus)[1]. And the RF power systems with 500MHz, 10kW klystron and $n_{\parallel}$(~4) variable comb-line antenna are being developed in collaboration with Korea Atomic Energy Research Institute and Kwangwoon university. The diagnostic tools such as magnetic probe and EBE (Electron Bernstein wave Emission) radiometer are being prepared to analyze the wave propagation and absorption characteristics via wave number and electron temperature. In 2016, fast wave coupling will be investigated to increase the coupling efficiency through the developed coupling model and edge density measurement of pre-ionization plasma produced by EBW heating and electron gun injection. Installation of RF systems and fast wave coupling experiments are planned in 2017 and the LHFW current drive experiments will be attempted in 2018.

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Thomson scattering process in turbulent plasmas

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The effects of turbulence on the Thomson scattering process are investigated in turbulent plasmas by the scattering of electromagnetic waves. The Thomson scattering cross section in turbulent plasmas is obtained by the fluctuation-dissipation theorem and plasma dielectric function as a function of the diffusion coefficient, wave number, and Debye length. It is demonstrated that the turbulence effect suppresses the Thomson scattering cross section. It is also shown that the turbulence effect on the Thomson scattering process decreases with increasing thermal energy. The dependence of the wave number on the total Thomson scattering cross section including the turbulent structure factor is also discussed.

Improvement of NBI heating efficiency with Magnetic field well

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Neutral Bean Injection (NBI) is one of main methods for plasma heating and current drive in magnetic fusion devices. Versatile Experiment Spherical Torus (VEST), recently constructed at Seoul National University [1] considers NBI as a main heating tool to access target advanced performance. However, large orbit loss is expected due to small toroidal field and plasma current in VEST. Sufficient beam power is difficult to be used, since high energy beam particles in a small field are suffering from large beam loss. In this study, we suggest adding a magnetic well structure to the low field side of the tokamak as a robust solution to improve such large beam losses. The effect of the well structure on reducing orbit loss is confirmed by a zero dimensional analysis [2, 3] and more detailed NUBEAM [4, 5] calculations. Finally, we adopt this well structure on VEST equilibrium to assess its effect. From NUBEAM simulations, we find that this magnetic field configuration can significantly improve the beam efficiency in VEST. This magnetic well configuration is envisaged to be applicable to other spherical tokamaks with the low magnetic field strength in enhancing the NBI efficiency.

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MAST Upgrade – Construction status and early research plans

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The MAST Upgrade programme research programme has three primary objectives: 1) To develop reactor-relevant advanced divertor concepts, 2) add to the knowledge base for ITER and 3) explore the feasibility of using a spherical tokamak as the basis for a fusion Component Test Facility.

To deliver this capability the load assembly is being comprehensively upgraded in stages and the first stage, known as "core scope", is now well into the assembly phase. Core scope includes 17 new shaping and divertor poloidal field coils (14 inside the vessel), and a new closed pump-able divertor structure to make a highly flexible exhaust physics platform. This stage of the upgrade will also provide a 50% increase in the toroidal field (from 0.585 (85kA) to 0.92 (133 kA) Tesla at R = 0.7m) and a near doubling of the inductive flux from the central solenoid (0.9 to 1.7Vs (1.6 Wb)), which should allow access to plasma current of 2MA. One of the present neutral beams will be moved off-axis for improved current profile control and fast ion physics studies. It will be equipped with an extensive gas fuelling system comprising 76 gas outlets allowing a good toroidal uniformity in the gas fuelling.

Many new diagnostics are included in the upgrade; in particular magnetic diagnostics for real time plasma position and shape control, as well as post pulse equilibrium reconstruction. The divertor has extensive Langmuir probe coverage, bolometry arrays, reciprocating probes, Thomson scattering, coherence imaging for divertor flows, imaging spectroscopy and infrared imaging.

Early research will focus on exhaust studies and specifically the production and control of various divertor configurations. The closed divertor will lead to lower main chamber neutral densities; as a result quantifying divertor closure, H-mode access and density-limits will be an early priority. The programme will expand to include advances in areas such as fast ion instabilities, core transport, pedestal and ELM physics.

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Study on non-inductive EBW heating experiment using direct XB mode conversion in VEST

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The Electron Bernstein Wave (EBW) can be effective heating method in Spherical Torus (ST) due to density limit according to low toroidal field. Since the direct XB mode conversion requires only a simple Low Field Side (LFS) waveguide for perpendicular injection compared to OXB mode conversion, the experiments of plasma generation and heating by using the direct XB mode conversion have been conducted in a small linear device with axial magnetic field bent and VEST (Versatile Experiment Spherical Torus). In the linear device, it is observed that over dense plasmas above X-mode cutoff density is generated by LFS X mode injection and the electron temperature peaks near Electron Cyclotron Resonance (ECR), indicating the presence of direct XB mode conversion and EBW collisional heating due to the short density scale length near edge Upper Hybrid Resonance (UHR) layer. In addition, the experimental results in the linear device for the effect of MicroWave (MW) multi-reflection using polarized slits reflecting X or O mode only show that the multi-reflected X wave can affect the direct XB mode conversion. The similar experimental result of XB mode conversion and EBW collisional heating are observed in VEST pre-ionization plasma that the pattern of density profile, the position and density scale length of mode conversion layer depend on the toroidal magnetic field and MW power. It is expected that those dependency on input MW power can be used for non-inductive EBW heating experiment in startup phase has been conducted for efficient XB mode conversion by controlling magnetic scale length (L_B) and density scale length (L_n). The feasibility of non-inductive EBW heating and current drive using direct XB mode conversion will be also investigated using 1D full wave mode conversion efficiency calculation code [1] and GENRAY ray tracing code.

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Plasma Start-up Experiment using Trapped Particle Configuration in VEST and KSTAR

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A mirror-like trapped particle configuration (TPC) has been applied to the ECH-assisted plasma start-up scenario in VEST by substituting the conventional magnetic field null configuration. An enhanced particle confinement of the pre-ionization plasma as well as the early and fast plasma current initiation have been observed with this TPC start-up scenario in VEST, due to the reduced open field particle loss by particle trapping and the radial/vertical stability of the current channels [1]. A relation is found between the initial poloidal field strength and the pre-ionized plasma density for successful plasma current initiation in various experimental conditions with 10 kW of 2.45 GHz ECH and the electron gun. Based on the result of VEST, TPC has been employed for the ECH-assisted plasma start-up scenario in KSTAR by overlapping the mirror like magnetic field on the conventional null configuration. A drastic enhancement of the open field confinement and the plasma current channel formation is also found with TPC even with the lower magnetic mirror ratio due to the larger aspect ratio of KSTAR compared with VEST. This result indicates that TPC can be used for improved plasma start-up in conventional tokamaks with a particular beneficial effect to the superconducting magnet operation. Moreover, TPC is expected to be helpful for the solenoid free start-up using the outer PF coils only, which generally requires the large pre-ionization power due to large stray fields.

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Electron-ion bremsstrahlung process in turbulent plasmas

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The effects of plasma shielding and turbulence on the electron-ion bremsstrahlung spectrum are investigated in turbulent plasmas. The effective potential taking into account the plasma shielding and the plasma turbulence through the diffusion effect with the impact parameter analysis is employed in order to obtain the bremsstrahlung radiation cross section as a function of the Debye length, diffusion coefficient, impact parameter, projectile energy, photon energy, and thermal energy in turbulent plasmas. It is found that the bremsstrahlung radiation cross section decreases with increasing thermal energy for small impact parameters and, however, increases with an increase of the diffusion coefficient. It is also found that the plasma shielding effect enhances the bremsstrahlung radiation cross section for small thermal energies. In addition, it is found that the plasma shielding effect on the bremsstrahlung cross section in turbulent plasmas increases with an increase of the diffusion coefficient and, however, decreases with an increase of the thermal energy.

Analysis of fast-ion D_α data from the National Spherical Torus Experiment

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Abstract

Measured fast-ion D_α (FIDA) data from an extensive NSTX database are compared to "classical" predictions that neglect transport by instabilities. Even in the absence of detectable MHD, in virtually all cases, the profile peaks at smaller major radius and the profile is broader than the predictions. Abrupt large-amplitude MHD events flatten the FIDA profile, as do most toroidal Alfvén eigenmode (TAE) avalanche events. Generally, the onset of a long-lived mode also flattens the FIDA profile. There is a shortfall of high-energy ions at large major radius in discharges with repetitive TAE bursts. If available, preliminary FIDA data from NSTX-U will also be presented.

Mitigation of ELM peak heat loads on NSTX-U through impurity granule injection

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Controlling the peak heat load associated with an Edge Localized Mode (ELM) is especially crucial in Spherical Tokomaks where the small major radius provides minimal opportunity for radial relaxation of the heat profiles prior to connection with the divertor. Thus it is critical to develop a method for mitigation of these events to ensure stable operation of next generation ST devices. As it has been shown that there is an inverse relationship between ELM frequency and the peak heat flux delivered during the mode, a system has been developed for NSTX-U whereby ELMs will be paced at a rate 10+ times higher than the natural ELM frequency by injection of impurity microgranules into the edge plasma. Granules of low Z impurity species (Li, B, C) are radially driven into the midplane edge of the discharge through impact acceleration with a rapidly rotating impeller. The rotation speed of the impeller determines the granule injection velocity within a range of 50 – 150 m/sec. In addition, the impeller frequency, coupled with the input rate of the granules, sets the overall particle injection frequency at up to 200 Hz. The granules, upon impact with the edge plasma, ablate and generate an overdense flux tube within the H-mode pedestal. This leads to a ballooning instability, resulting in the production of an ELM. These paced ELMs are then able to regulate the pedestal in a controlled manner, moderating the peak heat flux to a level tolerable to the plasma facing components. The mass deposition and ablation model benchmarked through lithium granule injection on DIII-D and its application to the plasma edge parameters found on NSTX-U will be discussed along with an outline of the upcoming NSTX-U experiments.

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Effect of Lithium on the Edge Plasma in NSTX and NSTX-U

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Strategic planning studies have identified control of the plasma-material interface as a critical area for realization of power production. Solid plasma-facing components (PFCs) are the leading candidates for future devices, predominantly serving as PFCs for present devices. While ITER has been designed to work with W and Be, there is little safety margin on heat flux removal capability[1]. The power exhaust challenge for reactors the size of ITER is even harder, requiring higher amounts of core and divertor radiation[2]. Liquid metal PFCs have some attractive features that could remove some of the restrictions of solid PFCs. The typical erosion and PFC performance degradation of solid PFCs can be obviated with self-healing surfaces; the challenge shifts to controlling core impurity content, and managing tritium retention. Finally lithium PFCs, both solid and liquid, can provide access to low recycling, high confinement regimes, e.g. at $\geq$2x H-mode scalings[3], enabling attractive core and edge plasma scenarios.

Lithium coatings were used to reduce recycling, improve edge plasma performance and eliminate ELMs in NSTX[4-6], owing to the inward shift of the density and pressure profiles. Recent analysis confirmed that these profile and stability effects were also present in highly-shaped, high-performance NSTX discharges[7]. A liquid lithium divertor demonstrated control of lithium inventory, and prevented droplet injection via droplet size control[8]. Lithium coatings are the first stage of the liquid metal PFC program in NSTX-U[9], and results of a controlled lithium dose scan will be reported, if available, for comparison with previously published NSTX results.

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Results from LTX with solid and liquid lithium walls

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The Lithium Tokamak Experiment (LTX) is an ohmically heated moderate-sized low aspect ratio tokamak with a heated liner or shell, which covers 80 % of the plasma surface area (4 m^2). In 2014, a new approach to lithium wall coatings was developed. Two 1.5 – 2 kW electron beam systems are used to evaporate lithium from a pool of liquid at the bottom of the lower shell. The e-beam system produces a 50 – 100 nm coating of liquefied lithium on the heated shells in < 5 minutes. Discharges using the new evaporative coating system have strongly reduced impurities, especially oxygen. Confinement in LTX ohmic discharges is now improved by up to 10×, compared to previous results with helium-dispersed coatings. Confinement times now exceed ITER ELMy H-mode scaling by 2-4× [J. C. Schmitt et al., Phys. Plasmas **22**, 056112 (2015)], with liquid or solidified lithium coatings. Core impurity concentrations of lithium have been measured to be <0.5%, with full liquid lithium walls at 240 C. This is the first experimental evidence that high performance tokamak discharges are compatible with large-area liquid lithium walls. Surface analysis of the e-beam evaporated films has been performed, which shows that the surface is primarily clean metallic lithium. Recent experiments in which the discharge density is fueled by high field side gas puffing to ~3×10^{19} m^{-3}, and then allowed to decay with no further fueling, show the development of broad, flat electron temperature profiles to the last closed flux surface, after gas puffing is terminated. Broad, flat electron temperature profiles have long been predicted as a consequence of very low recycling walls [S.I. Krasheninnikov, L. E. Zakharov, and G. V. Pereverzev, Phys. Plasmas **10**, 1678 (2003)], but this is the first observation of such profiles. Preliminary results for lithium concentrations in the core plasma with this very hot (>200 eV) edge are in the 5-10% range. Results and near term plans for an upgrade of LTX (LTX-U) will be summarized.

Experimental investigation of magnetic-field topology via perturbation method in the PFRC-2 device

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The fundamental question about FRC experiments is whether an FRC has actually formed; specifically, does there exist a magnetized, high-beta, plasma configuration with both no toroidal field and a simply connected configuration-space separatrix? We have investigated the latter part of this question, the existence of a separatrix, in the RMF_o-driven PFRC-2 device. The method involved applying an externally generated periodic RF power perturbation to open-field-line plasma at one (remote) axial end of the device while simultaneously searching for evidence of this periodic perturbation near the presumptive FRC. Measurements of the floating potential and ion saturation current were taken at the axial center of the device using a cylindrical Langmuir probe (r = 0.025 cm, L = 0.20 cm). When applying no RMF_o, measurements of the floating potential and the ion saturation current showed evidence of the frequency signature of perturbation throughout the entire radial profile of the device, indicating open field lines throughout the device. However, under certain conditions when operating the RMF_o, we found no evidence of the perturbation within a certain radius, but instead see the perturbations only beyond this particular radius, indicating a separatrix inside which the magnetic field lines are closed. Analysis of the precise nature of the perturbation within the open field plasma as well as the range of operating conditions during which evidence of a separatrix has been found are presented.

Maximising fusion power density in NBI driven systems

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For a number of fusion applications such as component test facilities and fission-fusion hybrid schemes, fusion power density, not energy multiplication, is the primary measure of performance. It is well established that the fusion power density can be raised above the thermonuclear level via NBI and previous studies [*Jassby, D. L, Nuclear Fusion 15(3) 1975*] have shown that at fixed plasma pressure the fusion power density is inversely related to the energy multiplication factor, *Q*. However, this increase in fusion power density is driven by the increase in the NBI power required to sustain the plasma pressure. In all fusion devices, and particularly in compact STs, the applied power is strictly limited by the allowable internal wall and divertor power loadings. Therefore it is instructive to fix the NBI power and determine the effects of confinement on fusion power density. Both pure deuterium and deuterium-tritium systems are investigated, with discussion focused on practical devices conceivable with current technologies. It is demonstrated that, contrary to the fixed pressure case, under the constraint of fixed beam power fusion power density increases with *Q*.

Implementation of a 3D halo neutral model in the TRANSP code and application to projected NSTX-U plasmas

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A 3D halo neutral code developed at the Princeton Plasma Physics Laboratory and implemented for analysis using the TRANSP code is applied to projections of plasma performance for the National Spherical Torus experiment-Upgrade (NSTX-U). The 3D halo neutral code uses a "beam-in-a-box" model that encompasses both injected beam neutrals and resulting halo neutrals. Upon deposition by charge exchange, a subset of the full, one-half and one-third beam energy components produce first generation halo neutrals that are tracked through successive generations until an ionization event occurs or the descendant halos exit the box. The 3D halo neutral model and Neutral Particle Analyzer (NPA) simulator in the TRANSP code have been benchmarked with the Fast-Ion D-Alpha simulation (FIDAsim) code, that provides Monte-Carlo simulations of beam neutral injection, attenuation, halo generation and halo spatial diffusion. When using the same atomic physics database, FIDAsim and TRANSP simulations get excellent agreement on the spatial profile and magnitude of beam and halo neutral densities as well as the NPA energy spectrum. The simulations show that halo neutrals remain in the vicinity of the neutral beam footprint as expected and that halo neutral density can be comparable with beam neutral density. The halo neutrals can double the NPA flux, but have minor effects on the shape of the NPA energy spectrum.

Overview of Research Plans for NSTX Upgrade*

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The missions elements of the National Spherical Torus eXperiment Upgrade (NSTX-U) research program are to: (1) establish the physics basis for the spherical tokamak (ST) as a candidate for a Fusion Nuclear Science Facility (FNSF), (2) understand and develop novel solutions to the plasma-material interface (PMI) challenge, and (3) advance the understanding of toroidal confinement physics for ITER and beyond. Underlying all of these missions is access to a unique plasma physics parameter regime of high beta combined with reduced collisionality to address fundamental questions about plasma stability and turbulent transport, and greatly extend understanding of toroidal plasma science. To achieve this collisionality reduction with equilibrated profiles, NSTX-U will double the toroidal field, plasma current, and NBI heating power relative to NSTX and will also increase the pulse length from 1–1.5 s to 5–8s. NSTX-U achieved first test plasma in August of 2015, and the NSTX Upgrade Project is now complete. In preparation for utilizing the new capabilities of NSTX-U, the NSTX team completed a comprehensive 5 year research plan in 2013, and the major directions of experimental research for the next three years are well defined. Major elements of the plan include assessments of more tangential injection of the 2nd NBI of NSTX-U for increasing the NBI current drive by up to a factor of 2 and supporting 100% non-inductive operation. The team will also assess NBI plus bootstrap current over-drive for providing non-solenoidal current ramp-up as needed for an ST-based FNSF. NSTX-U researchers will explore novel solutions to the power exhaust challenge for FNSF and DEMO by testing partial detachment, extreme flux expansion using a snowflake or X-divertor, and by testing liquid metal plasma facing components (PFCs). In support of ITER and FNSF, NSTX-U researchers plan to develop advanced disruption avoidance and mitigation techniques and predictive capability for non-linear Alfven eigenmode “avalanches” which can expel fast-ions in NSTX and may exist in the ITER hybrid and reversed shear scenarios. Research plans for a range of topical science areas will be described.

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Configuration Studies for Next-Step Spherical Tokamaks*

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The spherical tokamak (ST) is a leading candidate for a Fusion Nuclear Science Facility (FNSF) due to its potentially high neutron wall loading and modular configuration. Possible FNSF missions include: providing high neutron flux (1-2MW/m^2) and fluence (3-6MWy/m^2), demonstrating tritium self-sufficiency (tritium breeding ratio TBR $\geq$ 1), and demonstrating electrical self-sufficiency. All of these missions must also be compatible with a viable divertor, first-wall, and blanket solution. Recent U.S. studies have for the first time developed ST-FNSF configurations simultaneously incorporating: (1) a blanket system capable of TBR ~ 1, (2) a poloidal field (PF) coil set supporting high κ and δ for a range of l_i and β_N values consistent with NSTX/NSTX-U previous/planned operation, (3) a long-legged / Super-X divertor analogous to the planned MAST-U divertor which substantially reduces projected peak divertor heat-flux, (4) all outboard PF coils outside the vacuum chamber and superconducting to reduce power consumption, and (5) a vertical maintenance scheme in which blanket structures and the centerstack (CS) can be removed independently. High-temperature superconducting (HTS) magnets are also potentially attractive for compact ST applications due to higher operating temperature, which could reduce cryogenic load requirements and overall device size relative to configurations that utilize low-temperature superconductors (LTS). HTS conductors can also operate with very high current densities and high magnetic fields. Recent studies have shown that with only a modest central solenoid, the optimal aspect ratio for a HTS tokamak pilot plant is between A = 1.7 and 2.3 depending on inboard shielding thickness. These results point to the interesting finding that the optimal aspect ratio for a compact HTS pilot plant may be near A = 2, which is an unexplored configuration in the present fusion program.

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Perspectives of the PROTO-SPHERA experiment

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The scientific objectives of the PROTO-SPHERA experiment, that aims to replace the metal center post of a Spherical Torus (ST) with a Screw Pinch plasma (SP), can be summarized as follow: 1) build up the ideal MHD stable ST+SP configuration destabilizing the plasma central column (raising its current I_e from 8.5 kA to 60 kA in less than 1 ms) and "tunneling" through a non-linear unstable kink; 2) compress the ST to the lowest possible aspect ratio A=R/a, with an elongation $\kappa>2$, by a fast rise (0.5-1 ms) of the current in the PF compression coils, producing in this way a Flux Core Spherical Tokamak rather than a Flux Core Spheromak already obtained in TS-3 or Sphex; 3) sustain the coupled configuration by Helicity Injection (HI) from the central plasma column – fed by annular electrodes – to the ST (with a toroidal current up to I_p=240 kA), through resistive instabilities at the torus edge, exceeding at least one resistive time-scale (~70 ms); 4) compare the ST energy confinement time τ_E with the one of conventional Spherical Tokamaks of similar size (e.g. START), in order to assess any possible degradation due to HI.

The first phase of PROTO-SPHERA is producing the central column plasma only. Its first results have already removed the major concern that the Screw Pinch plasma could attach itself on a restricted portion of the annular anode, due to the unexpected and favorable effect of the **E∧B** drift. The major components to be built in order to complete the PROTO-SPHERA machine are: 1) the variable current PF compression coils that are more sophisticated than the already built constant current PF shaping coils, due to the fast current rise required for the ST formation; 2) an extension from 54 to 324 cathode emitters in order to drive I_e=60 kA; 3) the upgrading of the SP plasma power supply and the new variable current PF compression coils power supply. The recent development of the so-called Supercapacitors can allow to design in a robust and simple way the power supplies required for the fast rise of the Screw Pinch current and the compression PF coils currents, overcoming the overshoot and feedback problems inherent in such demanding components.

In conclusion, at this moment it seems possible to overcome all the most critical technical points that could have hindered the final completion of PROTO-SPHERA machine.

Development of Over 1 MW and Multi-Frequency Gyrotrons for Present ST and Fusion Experiments in University of Tsukuba

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The development of wide frequency range from 14 to 300 GHz of high power mega-watt gyrotron for fusion is in progress in University of Tsukuba. The strong development activity was carried out in collaboration with JAEA, NIFS, TETD and universities. Over-1 MW dual frequency gyrotron of new frequency range (14 - 35 GHz) has been developed for EC/EBW H&CD for GAMMA 10/PDX, QUEST, NSTX-U and Heliotron J. Output power of 1.25 MW at 28 GHz and estimated oscillation power of 1.2 MW at 35.45 GHz from the same tube have been achieved with the cathode angle improvement and two frequency window. This is the first demonstration of the over 1 MW dual-frequency operations in lower frequency, which contributes to the technology of wide band multi-frequency/multi-MW tube. The output power of 600 kW for 2 s at 28 GHz is also demonstrated. It is applied to the QUEST and has resulted higher EC-driven current than ever. Further development of dual-frequency gyrotron with 2 MW 3 s at 28 GHz, especially designed for G-10/PDX and NSTX, 0.4 MW-CW for QUEST, and power exceeding 1 MW for 3 s at 34.8 GHz for Heliotron J have been started. As for higher frequency range, a new frequency of 154 GHz has been successfully developed, which delivered 1.16 MW for 1 s and the total power of 4.4 MW to LHD plasma with other three 77 GHz tubes, which extended the LHD plasma to high T_e region. All these gyrotron performances are new records in each frequency range. The sub-THz gyrotron development is also just begun in collaboration with JAEA for Demo-Reactor ECH system. In the preliminary experiment, the 520 kW power was obtained at 299.85 GHz.

Studies of Transient CHI Plasma Start-up on HIST

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An advantage of the Spherical Torus (ST) is the low aspect ratio and so elimination of a central solenoid coil is required for attractive high-beta fusion reactors based on the ST concept. Thus alternate methods for the plasma start-up that do not rely on the central solenoid are necessary for the viability of the ST concept. The transient coaxial helicity injection (T-CHI) without requiring for dynamo is a promising candidate for the non-inductive plasma start-up. So far, the T-CHI method has been successfully applied to NSTX for the start-up followed by inductive ramp-up. This coupled discharge has now achieved plasma currents larger than 1 MA [1].

We have recently examined the T-CHI method on the Helicity Injected Spherical Torus (HIST) device [2]. Understanding the physics of the flux closure during T-CHI still remains as a key issue, which is the primary purpose of the experiment on HIST. In the experiment, the internal magnetic field measurements (2D flux plots) have verified the plasmoid injection and the following formation of the closed flux surfaces (flux closure) during the start-up phase. The formation of an X-point after bubble burst has been generated by fast magnetic reconnection event. The MHD simulation [3] on T-CHI suggests that the faster reconnection times compared to the resistive time are consistent with the Sweet-Parker model. The closed poloidal flux increases as increasing the injection voltage. However, after the formation, the current injected continuously from the gun distorts helically the open field lines due to onset of the n=1 kink instability. The electron density decreases rapidly as the instability is triggered. The plasma starts to decay after the injection current terminates. During the decaying phase, the inner edge current diffuses toward the magnetic axis and the magnetic configuration relaxes to the axisymmetric state. The current density profile can be controlled by optimizing the bias (injector) flux so that the kink instability does not occur. The ion and electron temperatures have been investigated by ion Doppler spectroscopy (IDS) and double electrostatic probe measurements.

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Polarization control of incident microwave for non-inductive formation of spherical tokamak by electron Bernstein wave heating and current drive in LATE

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Non-inductive formation of spherical tokamak in LATE is performed by electron Bernstein (EB) wave heating and current drive. The microwaves at 2.45GHz from four magnetrons are injected obliquely to the toroidal field from low field side on mid-plane.

Line-averaged electron density increases up to about seven times the plasma cutoff density at the last stage of a discharge, when the fundamental electron cyclotron resonance (ECR) layer is located in the plasma core and the second ECR layer is located outside the upper hybrid resonance (UHR) layer. It is important to control the polarization of incident microwave in order to optimize the mode-conversion rate from incident microwave to EB wave.

The optimal injection polarization varies from O-mode like polarization to X-mode like one as the electron density gradient near the UHR layer becomes higher, according to the linear mode-conversion rate theory with cold plasma resonance absorption model in a slab geometry [1].

Two polarizers are used to convert the rectangular TE_{10} mode from magnetrons to O-mode like polarization or to X-mode like one [2]. The powers of the O-mode like polarization and the X-mode like one are changed by preprogrammed control, respectively. The driven plasma current is increased by 20% in the case that the power fraction of the O-mode like polarization is larger than that of the X-mode like one at the first stage of a discharge and then that of the X-mode like one is larger than that of O-mode like one at the last stage of a discharge, compared to the case that all the injection power is O-mode like one.

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High Power Heating of Magnetic Reconnection in ST Merging Experiments: TS-3, TS-4, UTST and MAST

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Since 1985, we have been investigating toroidal plasma merging and reconnection for high-power heating of spherical tokamak (ST) and field-reversed configuration (FRC), using TS-3 (ST, FRC: R~0.2m, 1985~), TS-4 (ST, FRC: R~0.5m, 2000~), UTST (ST: R~0.45m, 2008~) and MAST (ST: R~0.9m, 2000~) devices. Our merging experiments realized a significant reconnection heating over 1.2keV in the world-largest ST merging experiment: MAST [1,2] after detailed 2D elucidation of ion and electron heating up to 250eV in TS-3 and TS-4 ST merging experiments [3,4]. They revealed clear energy-conversion mechanisms of magnetic reconnection: huge outflow heating of ions in the downstream and Ohmic heating of electrons around the X-point. The reconnection outflow accelerates ions up to 70-80% of Alfven speed of reconnecting magnetic field, and they are thermalized by fast shock-like density pileups in the downstreams. The series of experiments agree that the reconnection heating energy is proportional to square of the reconnecting magnetic field. The guide toroidal field does not affect the bulk heating of ions and electrons, probably because the reconnection/ outflow speeds are determined mostly by the externally driven inflow by the help of several fast reconnection mechanisms. Their detailed mechanisms were further investigated in collaboration with the Hinode satellite observation of solar coronal heating and with various PIC simulations. Those physics, particularly the reconnection heating and acceleration lead us to construction of the new up-graded high magnetic field merging experiment: TS-U (2016~) in University of Tokyo. This talk reviews major progresses in those international and interdisciplinary studies of toroidal plasma merging experiments for physics and fusion applications of magnetic reconnection.

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Status of non-inductive startup via the electrostatic helicity injection in Versatile Experiment Spherical Torus

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The electrostatic helicity injection is one of the powerful methods for non-inductive start-up of spherical torus. By using this methods, it is possible to expand startup capability of Versatile Experiment Spherical Torus (VEST). The injector (electron gun) has been installed in the lower chamber of VEST and experimental attempts to drive plasma current via the electrostatic helicity injection during plasma startup are underway.

By using single power configuration that can provide voltages for both injector operation and helicity injection [1], gradual rising plasma current of ~ 10kA has been achieved successfully based on injection current of 1.5kA and the injection voltage of 500V. The measured time evolution of plasma current density distribution using magnetic diagnostic system of VEST and the fast camera images show the formation of current sheet clearly. The experiments with two power system also conducted to control injector operation and helicity injection respectively. Plasma currents up to ~30kA have been achieved based on the peak injection current of 1.0 kA and the peak injection voltage of 1.5 kV. The large current multiplication is also confirmed, further experiment and more accurate diagnostic is ongoing to confirm the possibility of relaxation into tokamak-like plasma with closed flux surface.

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Field-Aligned SOL Losses of HHFW Power and RF Rectification in the Divertor of NSTX

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The National Spherical Torus eXperiment (NSTX) can exhibit a major loss of high-harmonic fast wave (HHFW) power to the upper and lower divertor regions along scrape-off layer (SOL) field lines passing in front of the antenna, resulting in bright and hot spirals on both the upper and lower divertor regions. This appears to be a fast-wave propagation effect, but it is not clear whether it is general to fast-wave systems or specialized to the HHFW regime or to spherical-tokamak geometry [1]. One possible mechanism for converting wave power into a heat flux is RF sheaths forming at the divertors. We present swept-voltage Langmuir probe characteristics for probes under the spiral that are shifted relative to those not under the spiral in a manner consistent with RF rectification. From this shift in characteristics, we estimate both the magnitude of the RF voltage across the sheath and the sheath heat flux transmission coefficient in the presence of the RF field [2]. Although precise comparison between the computed heat flux and infrared (IR) thermography cannot yet be made, the computed heat deposition compares favorably with the projections from IR camera measurements. The RF sheath losses, calculated in this fashion, are significant and contribute substantially to the total SOL losses of HHFW power to the divertor for the cases studied. This work will guide experimentation on NSTX-U, where a wide-angle IR camera and a dedicated set of coaxial Langmuir probes for directly measuring the RF sheath voltage will quantify the contribution of RF sheath rectification to the heat deposition from the SOL to the divertor.

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Energetic Particle research on NSTX-U

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Spherical Torii with Neutral Beam (NB) injection typically feature a large fraction of super-Alfvénic fast ions. Those energetic particles (EP) can drive a number of instabilities, which in turn affect the EP population and related quantities such as NB heating and current drive. The resulting deterioration of plasma performance must be addressed while moving forward with the ST programme, e.g. towards a ST-based Fusion Nuclear Science Facility. Experiments on NSTX have resulted in an extensive characterization of EP-driven instabilities over a broad range of frequency, including fishbones and other kink-like modes at low frequency (10's of kHz), toroidal Alfvén eigenmodes (TAEs, near the Alfvén frequency) and Global/Compressional AEs near the ion cyclotron frequency. The effects of those instabilities have been measured, indicating – for example – the deleterious effect of large-amplitude, bursting TAE events on EP confinement. This work will present the latest results from analysis of NSTX data, including modeling of mode stability through linear and non-linear codes, and modeling of the associated fast ion transport. In particular, the development and validation of reduced models for fast ion transport by AEs will be emphasized, focusing on the improvement of predictive capabilities for EP physics that is relevant for both STs and conventional aspect ratio tokamaks. The completion and start of operations of NSTX-Upgrade offer enhanced tools to further improve our understanding of EP physics, e.g. higher toroidal field and plasma current, and an additional set of NB injectors. An overview of the planned experiments for the coming NSTX-U campaign will be given.

Towards fully non-inductive operation in NSTX-U

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The start-up, ramp-up, and sustainment of a tokamak plasma utilizing little to no induction from a central solenoid is a major challenge in magnetic fusion. Because of the scientific and operational challenges, the problem is best solved by employing an iterative loop between experiment and simulation where validation plays a critical role in the improvement of modeling and in the projection to new experiments. On NSTX-U, high harmonic fast waves (HHFW) and NBI are available and can be combined to ramp the plasma current non-inductively. This work discusses time-dependent simulations and challenges in optimizing the non-inductive fraction in the early ramp-up phase. It is shown that only a combination of RF and NBI can bridge the phase between startup plasma (either ohmic or non-solenoidal) and ramp-up. In fact, the NBI is lost for shine through in the low density, start-up plasma and cannot be used to drive current in the early phase of the discharge. HHFW prepares a target plasma where NBI can be injected with minimal losses. Current profile control is critical in order to attain the desired target and avoid the peaking of profiles at start-up and ramp-up that are conducive to ideal MHD instabilities.

An interesting synergy is observed between Electron Cyclotron waves and HHFW, depending on the phasing of the antenna. It is shown that the addition of EC wave heating can significantly increase the effectiveness of the RF power and relax the requirements on the total level of power that must be coupled to the start-up plasma. With 1 MW of EC power, the total power that needs to be coupled to the plasma to drive 300 kA of direct fast wave current is reduced from 4 MW to 1.5 MW. This work describes and revisits these simulations in light of the first experimental campaign on NSTX-U with emphasis on the limitations in our modeling capabilities.

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Coaxial Helicity Injection and Disruption Mitigation Studies in Support of NSTX-U, ST-FNSF, and ITER

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In support of solenoid-free plasma start-up for the ST, Transient Coaxial Helicity Injection (CHI) has been successfully used on both HIT-II and NSTX to generate substantial amounts of plasma current on closed flux surfaces. Numerous hardware upgrades to the NSTX-U device, that just became operational, will permit more than doubling of the plasma start-up current to over 400 kA, and this is supported by recent simulations using the TSC and NIMROD codes, which also show favorable scaling of the CHI start-up process with increasing machine size.

Application of transient CHI to future devices has been investigated through a conceptual design study of a CHI system for a ST-FNSF, which has resulted in two designs. The first configuration has features similar to the proven NSTX design. In this configuration, the entire blanket structure is electrically insulated from the rest of the vessel. In the second configuration, a toroidal ring electrode is mounted on top of the blanket structure and this is used for CHI discharge initiation. Both configurations support injector flux values equivalent for a 2 MA start-up, using less than 10% of the divertor coil currents. A near-term experimental test of the concept, for this second design, is planned on QUEST.

Predicting and controlling disruptions is an important and urgent issue for ITER. Methods to rapidly quench the discharge after an impending disruption is detected are also essential to protect the vessel and internal components of an ST-FNSF. In support of this activity, NSTX-U will employ three Massive Gas Injection (MGI) valves that are very similar to the double flyer plate design being considered for ITER. NSTX-U will be the first device to operate this valve design in plasma discharges. These valves have been tested off-line and deliver the required amount of gas (~ 200 – 400 Torr.L) to support NSTX-U experiments, which will offer new insight to the MGI data base by studying gas assimilation efficiencies for MGI gas injection from different poloidal locations, with emphasis on injection into the private flux region. The valve has also been successfully operated in external magnetic fields of 1 T.

While the MGI system may be adequate for most disruptions, the warning time for the onset of some disruptions could be much less than the MGI system response time. To address this important issue, a novel system based on the rail-gun concept has been designed, and plans for an off-line experimental test are in progress. The device referred to as an Electromagnetic Particle Injector (EPI) is fully electromagnetic, with no mechanical moving parts, which ensures high reliability after a period of long standby. In addition to responding on the required fast time scale, its performance substantially improves when operated in the presence of high magnetic fields. The system is also suitable for installation in close proximity to the reactor vessel.

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Studies of NSTX L and H-mode Plasmas with Global Gyrokinetic Simulation

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Plasma turbulence is considered one of the main mechanisms for driving anomalous thermal transport in magnetic confinement fusion devices. Based on first principle model, gyrokinetic simulations play an important role in studying the relation between plasma turbulence and anomalous thermal transport. In order to predict the confinement performance of future devices, it is crucial to validate gyrokinetic codes against experiments. Nonlinear local gyrokinetic simulations have been used to asses turbulence-driven transport in NSTX L and H-mode plasmas [1,2], and agreement in thermal transport with experiments has only been observed in limited cases. Due to the larger ρ^* of NSTX compared to conventional tokamaks, global effects may be important in determining thermal transport. Here, we present nonlinear global gyrokinetic simulations of NSTX L and H-mode plasmas using global gyrokinetic code GTS [3] and comparisons with experimental transport analysis. Comparisons with nonlinear local gyrokinetic simulations will be also be presented. The work is supported by DOE and computational resource is provided by NERSC.

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New Results from Non-solenoidal Startup via Local Helicity Injection on PEGASUS and Projections for NSTX-U

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Non-solenoidal plasma startup via local helicity injection (LHI) at the PEGASUS Toroidal Experiment now provides routine operation at $I_p \approx 0.17$ MA with $I_{inj} \approx 5$ kA and $V_{inj} \approx 1$ kV from four active arc injectors. Studies in the past year have advanced the understanding of the governing physics of LHI and its supporting technology. Experiments have yielded new insight into the injector impedance, which scales as $V_{inj}^{1/2}$ at useful LHI voltages and is governed by two effects: a quasineutrality constraint related to the tokamak edge density, and double-layer sheath expansion related to the injector arc density. Experiments varying the shape, $I_p(t)$, and helicity input test a predictive 0D power-balance model for LHI startup. These tests increase confidence in the underlying analytic low-A plasma inductance models used to predict the $I_p(t)$ evolution in PEGASUS plasmas and projections for NSTX-U. More sophisticated predictive modeling, however, requires accurate knowledge of the confinement properties during LHI and its scaling with B_T. Initial measurements of the electron temperature profile in LHI have been made with multi-point Thomson scattering to address this need. They indicate a surprisingly high $T_e > 300$ eV in the plasma core. This may indicate the dominance of high-energy electron fueling from the injector current streams and better-than-expected confinement in the core region, as the electron beam is confined long enough to thermalize. If verified, these measurements will impact the conceptual design for LHI on NSTX-U. Finally, the design of new divertor injectors for PEGASUS that increase available helicity injection to attain higher I_p and enable electron confinement studies in plasmas with strong helicity drive (like those projected for NSTX-U) will be discussed. Proposed upgrades to the PEGASUS facility will extend these LHI studies to higher B_T and I_p, enabling experiments at NSTX-U relevant parameters. This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences, under Award Numbers DE-FG02-96ER54375 and DE-SC0006928.

Features of the Rodless Ultra Low Aspect Ratio Tokamak

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The replacement of the conventional metal centre-post in spherical tokamaks (STs) for a plasma centre-post (PCP, the TF current carrier) is the ideal scenario for a ST reactor. A simple rodless ultra low aspect-ratio tokamak (RULART) using a screw-pinch PCP ECR-assisted with an external solenoid has been recently proposed, aiming to be the most compact RULART[Ribeiro C., SOFE-2015]. In that proposal, the solenoid provided the stabilizing field for the PCP and the toroidal electrical field for the tokamak start-up plasma, which, after evolving, will stabilize further the PCP, acting as stabilizing closed conducting surface. That RULART will require relative low TF, and the compactness (high ratio of plasma-vessel volume) may provide passive stabilization and easier access to L-H mode transition. New features of this RULART will be presented: 1) stability analysis of the PCP, which initially has a MHD stable hollow current profile; 2) equilibrium simulations for the tokamak plasma, and 3) its potential use with helium plasmas for assisting experiments of aneutronic reactions. This is envisaged via pairs of proton (p) and boron (^{11}B) ion beams, whose sources can be arranged in several ways, such as placed symmetrically (top & bottom) to respect of the vessel horizontal mid-plane (VHMP), at the HFS, with a quasi-vertical line-of-sight (sufficiently for the beams miss the sources of each other, while allowing a near maximum relative velocities, thus reactivity). The p-^{11}B reactions should occur at HFS close to the PCP surface, and between VHMP and the ring-type anode. Some born α-particles should cross the PCP and be dragged by the ion flow (higher momentum exchange) towards the anode but, unlikely this ion flow, will not bend towards the anode but escape directly into a direct electricity converter placed behind of it, since $v_{\alpha} >> v_{i\text{-}drift}$. The energy of the α-particles which fail to the reach the convertor will reach evenly the vessel directly or via thermal diffusion, after heating the plasma [favorable by the large excursion (~2a), if the α-particles are created at VHMP], leading to the lowest power wall load possible, because the spherical vessel. This might be a potential hybrid direct-steam cycle conversion reactor scheme, nearly aneutronic, and with no ash or particle retention problems, as opposed to the D-T thermal reaction proposals.

Experimental Study of Density Gradient Stabilization Effects on High-k Turbulence

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Theory and experiments have shown that electron temperature gradient (ETG) turbulence on the electron gyro-scale, $k_\perp \rho_e \leq 1$, can be responsible for anomalous electron thermal transport in typical NSTX H-mode plasmas. Electron scale (high-k) turbulence is studied in NSTX using a high-k microwave scattering system [Smith RSI. 2008]. We report on the stabilization effects of the electron density gradient on electron-scale density fluctuations in a set of neutral beam injection (NBI) heated H-mode plasmas. Density gradient was previously identified as a stabilizing mechanism of ETG turbulence in [Ren PRL. 2011]. The absence of experimental high-k density fluctuations is correlated with large equilibrium density gradient, consistent with linear stabilization of ETG modes by using the analytical ETG linear threshold [Jenko PoP. 2001]. The scattered power from electron-scale turbulence is anti-correlated with the equilibrium density gradient, suggesting density gradient as a nonlinear stabilizing mechanism of high-k turbulence. Larger equilibrium density gradient leads to higher values of the wavenumber corresponding to the maximum in the fluctuation spectrum and to a lower value of the plasma frame frequency of detected density fluctuations. Linear gyrokinetic simulations using GS2 show a clear correlation between the wavenumber value at the maximum linear growth rate and the local value of the electron density gradient. Higher values of the electron density gradient are also shown to reduce the value of the real frequency of instability from GS2. Nonlinear electron-scale gyrokinetic simulations show that high electron density gradient reduces electron heat flux and stiffness, and increases the ETG nonlinear threshold, reaffirming a nonlinear change in ETG turbulence and fluctuation spectrum from large equilibrium density gradient as predicted from previous ETG nonlinear simulations [Ren PoP. 2012, Guttenfelder NF. 2013].

Global MHD Mode Stabilization and Control for Disruption Avoidance

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The near-complete elimination of plasma disruptions in fusion-producing tokamaks is the present "grand challenge" for stability research. Meeting this goal requires multiple approaches, important components of which are prediction, stabilization, and control of global MHD instabilities. Research on the recently-operating NSTX-U starts a plan of synergizing these elements to make quantified progress on this challenge. Initial results from disruption characterization and prediction analyses describe physical disruption event chains in plasmas from the existing NSTX database. Analysis of NSTX and DIII-D experiments show that understanding of the stabilization of global modes is unified by kinetic RWM stabilization theory with an interesting complementarity: enhanced stability is dominated by precession drift in NSTX and bounce orbit resonances in DIII-D. Stability therefore depends on the plasma rotation profile. A model-based rotation profile controller for NSTX-U using both neutral beams and neoclassical toroidal viscosity is shown in simulations to evolve profiles away from unstable states. Active RWM control is addressed using dual field component sensor feedback and a model-based RWM state-space controller. Comparison of measurements and synthetic diagnostics is examined for off-normal event handling. A planned 3D coil system upgrade can allow RWM control close to the ideal n = 1 with-wall limit. The initial plasma reconstructions generated in the NSTX-U device will be shown, presenting the status, challenges, and advantages of reconstructing low aspect ratio equilibria for use in the stability and control calculations needed for disruption prediction and avoidance.

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An overview of the HIT research program and its implications for magnetic fusion energy

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The HIT research group reports evidence of pressure confinement in sustained spheromak configurations, quantitative agreement with Hall-MHD simulations, and a possible new pathway towards economical magnetic fusion energy. In HIT-SI (R_o = 0.34 m, a = 0.2 m), two inductive helicity injectors oscillating 90^o out of phase, continually drive edge currents and impose magnetic perturbations that provide sustainment via imposed-dynamo current drive (IDCD).[1] Experimental results[2] indicate the sustainment of a spheromak configuration without gross confinement-degrading instabilities that have plagued spheromaks previously;[3] an example derivation of the IDCD mechanism is provided.[4] The level of quantitative agreement between the experiment and Hall-MHD simulations with NIMROD and PSI-TET is presented, along with suggestions of next steps to improve agreement. Lastly, an IDCD-driven spheromak reactor concept (the dynomak) is presented that indicates cost-competitiveness with conventional energy sources.[5] The dynomak utilizes a molten salt (FLiBe) blanket system for first wall cooling, neutron moderation and tritium breeding. A tritium breeding ratio (TBR) of greater than 1.1 has been calculated using a Monte Carlo N-Particle (MCNP5) neutron transport simulation. High temperature superconducting tapes (YBCO) are used for the equilibrium coil set, though more conventional low-temperature superconductors could also be used. The limiting equilibrium coil set lifetime is at least thirty full-power years. The primary FLiBe loop is coupled to a supercritical carbon dioxide Brayton cycle due to attractive economics and high thermal efficiencies. With these advancements, an electrical output of 1000 MW is produced from a thermal output of 2486 MW, yielding an overall plant efficiency of approximately 40%. The overnight capital cost of this 1 GWe power plant is estimated to be $2.7 billion in 2013 USD.

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Measurement, Characterization, and Suppression of Instabilities in the PFRC-2 Device

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The Princeton Field-Reversed Configuration-2 (PFRC-2) device is an experiment to investigate aspects of FRC reactor design that include confinement, RF-heating, stability, fueling, ash-exhaust and power-handling issues. When exploring modes of operation, we observe several saturated instabilities, presenting as oscillations in plasma parameters measured by interferometry, Langmuir probes, high-speed visible light photography, and RF power coupling. These oscillations vary in relative amplitude from <1% (local and global) to >30% (global), and in frequency from 5kHz to hundreds of MHz. We present data showing that these oscillations can be either extinguished or dramatically suppressed by changing the background magnetic field, increasing the Rotating Magnetic Field (RMF) antennae power, and puffing neutral gas into the FRC region. In this presentation, the oscillations and thresholds implied by the above measures are characterized, and relevance to a potential FRC-based fusion power plant is discussed.

Non-inductive plasma start-up experiments on the TST-2 spherical tokamak

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Non-inductive plasma start-up and subsequent plasma current ramp-up by the lower-hybrid (LH) wave are being investigated on the TST-2 spherical tokamak at the University of Tokyo (R_0 = 0.36 m, a = 0.23 m). A newly-developed traveling wave antenna called the capacitively-coupled combline (CCC) antenna, installed on the low-field-side midplane, is being used to excite the LH wave. Plasma current ramp-up up to 20 kA has been achieved so far in the low toroidal magnetic field regime, B_t < 0.1 T. The current drive efficiency is observed to drop sharply just below the mode conversion density limit. This density limit increases approximately linearly with the toroidal magnetic field. Results of RF power modulation experiments indicate that a large fraction of energetic electrons generated by the LH wave are lost quickly. Full-wave modeling is being carried out to describe quantitatively the experimental results of LH current drive on TST-2. The maximum achievable plasma current is found to increase with the magnetic field. Further ramp-up to higher plasma currents would require operation at higher toroidal fields and/or an improved wave launching. An upgrade of the toroidal field coil power supply is in progress. A new top-launch CCC antenna is also being fabricated. The LH wave launched from the top of the plasma undergoes a more favorable $n_{\parallel}$ up-shift and results in an almost complete absorption during its first pass through the plasma. This is expected to avoid parasitic dissipation of the LH wave in the scrape-off layer plasma. The achievement of higher plasma currents is expected to contribute to an improved current drive efficiency by providing better confinement of energetic electrons.

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Recent Progress in the SUNIST Spherical Tokamak

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An improved repeatability of the ohmic plasmas has been achieved after studying the temporal evolution of the gas pressure in the vacuum vessel and the dependence of the repeatability of plasma discharges on different timing sequences between the gas puffing pulse and the onset of the ohmic field. The vacuum vessel of SUNIST was split into two insulated hemispheres, both of which were insulated from the central cylinder. The eddy currents flowing in the vacuum vessel (VV) were modeled and experimentally measured. A 3D finite elements model indicated that when the poloidal field (PF) was applied, the induced eddy currents on the top and bottom of vacuum vessel had the same direction as the current flowed in the PF coils. These features resulted in the leading phases of signals on the top and bottom flux loops when compared with the PF waveforms. A magnetic probes array based on flexible printed circuit boards was mounted on the surface of the VV to measure the eddy current flowing in the wall of VV. The measurements confirmed the modelling results. A prototype of a super-fast reciprocating (up to 20 m/s) probe has been developed for SUNIST and makes it possible to get the radial profile of electrostatic parameters of the short pulse (< 20 ms) ohmic plasmas in one shot. Besides the technical progress, some physical results, including toroidal Alfven eigenmodes (TAE) excited by runaway electrons and edge plasmas properties, will also be presented.

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Application of merging/reconnection heating for spherical tokamak in MAST

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The world largest application of merging/reconnection heating, which was developed in START [1] and TS-3 [2], has been studied in detail using 130 channel YAG- and 300 channel Ruby-Thomson scattering measurement and a new 32 chord ion Doppler tomography diagnostics [3] in MAST. In addition to the previously achieved record heating of ~1keV [4], it is found that magnetic reconnection mostly heat ions globally in the downstream region and electrons locally in the X point by 2D imaging measurement of ion and electron temperature profile [5]. Electrons mostly gain energy around X point with the characteristics scale length of 0.02-0.05m < c/ω_{pi}, while ion temperature increases inside the acceleration channel of reconnection outflow with the width of c/ω_{pi} ~ 0.1m and the downstream where reconnected field forms thick layer of closed flux surface. The toroidal guide field mostly contributes to the formation of a localized electron heating structure at the X point and not to bulk ion heating downstream. The global reconnection heating of ions increases as a function of poloidal magnetic field B_p^2 by outflow heating mechanism. In the millisecond time scale startup experiment in MAST, the energy relaxation time between ions and electrons is in the comparable time scale (τ^E_{ei} ~ 4-11ms) and electrons are also heated globally after the delay of τ^E_{ei}, forming triple peak structure with the hot spots at the X point and downstream both with and without the assist of centre solenoid.

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Recent results on non-inductive formation of spherical torus by electron Bernstein wave heating and current drive in LATE

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The LATE device explores key issues on non-inductive start-up and formation of spherical torus by electron Bernstein (EB) wave heating and current drive. By injecting 2.45 GHz microwave from outboard side with an oblique angle to the toroidal field and locating the upper hybrid resonance (UHR) layer in the inboard side of the second electron cyclotron resonance (ECR) layer, the plasma current is ramped up with seven times the plasma cutoff density. This suggest that excitation of EB wave in the first propagation band is essential to heat the bulk plasma at the fundamental ECR.

Recently, the power supply for the toroidal coil has been refurbished to increase the maximum coil current and the same ECR condition for 5 GHz microwave as for 2.45 GHz one becomes available. The circulator for 5 GHz microwave injection system has been replaced and the microwave power injection from the klystron has been tested up to 190 kW for 70 ms. In the first experiment with 5 GHz microwave, the line-averaged electron density increased up to $\bar{n}_e \sim 1 \times 10^{12}$ cm^{-3}, which is ~2.5 times the case in the second ECR heating and current drive.

Heavy ion beam probe (HIBP) system has been developed to measure the space potential distribution. The primary Rb^+ beam with energy up to 20 keV is injected from the top port through the poloidal sweeper and the secondary beam coming to the bottom port is guided through the poloidal deflector to the energy analyzer. Preliminary result shows that the space potential is positive in the core plasma (~50 V) when Ip ~ 6 kA and $\bar{n}_e \sim 4 \times 10^{11}$ cm^{-3}.

Infrared measurements of divertor heat loads during steady state and transient events on MAST

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The power load to the divertor is a key challenge for future devices, such as ITER and DEMO, not only during transient events but also during steady state conditions. Measurements of the power load to the divertor have been performed in MAST using infrared (IR) thermography. The IR system is able to measure at spatial resolution of the order 1.5 mm and with temporal resolution of up to 7.5 kHz. In this work IR measurements in L mode, H mode and during ELMs are reported. Recent measurements of the SOL fall off length have predicted narrow widths for ITER. MAST data has been analysed in both L and H mode to investigate the parametric scaling of the fall off length in an ST. The key scaling quantities are found to be consistent with conventional aspect ratio devices, with a major dependence on the plasma current. The MAST data is consistent with the multi-machine scalings for ITER and predicts an H mode fall off length of 5.1 mm for MAST-U, however, this is likely to be broadened by the power spreading seen at low target electron temperatures. IR measurements in L mode show the filamentary nature of the SOL. The filaments at the divertor show that the radial filament size is of the order 5 mm. The measured target radial width maps, via the equilibrium magnetic field, to a toroidal size at the midplane of order 5 cm. The mapped size is consistent with estimates from visible imaging. The L mode filaments are seen to dominate the outer part of the heat flux profile to the tile, but carry only a small amount of power. ELMs are clearly filamentary in nature. Here, analysis of the heat loads during ELMs and the impact of ELM mitigation via resonant magnetic perturbation (RMPs) are reported. RMPs produce ELM mitigation with the peak heat flux halving for a halving of the ELM energy. Strike point splitting due to RMPs is also seen and measurements compared to predictions from plasma response modelling using MARS-F. IR measurements on MAST-U will be key in assessing the reduction of the divertor power load from the super-X divertor. The design and capabilities of the MAST-U IR system will be outlined, along with plans for future expansion.

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Structural Assessments of Magnets for the Next Generation Spherical Tokamaks

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Spherical tokamaks (ST) typically have lower fields which reduce the wedging pressures in the TF inner leg. The smaller radial build of the central column also reduces the wedging stress. If a conventional multiple coil case arrangement is chosen rather than the large single central conductor, then the out-of-plane (OOP) load on the TF inner leg must be taken by friction or mechanical keys. With the lower wedge pressure, friction can be a marginal torsional shear support mechanism. Advanced divertors will impose different out of plane loading and may introduce a different regime of OOP loading. ST's offer little space for a solenoid and inner corner shaping coils and will pose new PF coil support challenges. Structural analysis of 2 and 3 meter major radius next generation ST's is presented. The 3 meter design uses a proposed long legged super X configuration. Both TF and PF coils are evaluated. The TF coils are cased coils with HTS superconductor winding packs. Space allocation issues for the TF inner leg are also discussed. Structural contributions from the tape structure of the high temperature superconductor are considered.

Electron temperature and density profile measurement on the TST-2

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Non-inductive plasma current start-up experiments using the lower hybrid wave (LHW) have been performed on the TST-2 spherical tokamak. Profiles of the electron temperature T_e and density n_e were measured by Thomson scattering (TS) by accumulating many TS signals from reproducible discharges with plasma current $I_p \sim 5$ kA started-up and sustained by LHW power of 20 kW. The T_e profile was hollow while the n_e profile was centrally peaked. T_e, n_e and the electron pressure p_e at the plasma center were 6 eV, 6×10^{17} m^{-3}, and 0.6 Pa, respectively. On the other hand, the total pressure at the plasma center including the fast electron pressure and the ion pressure, deduced from EFIT equilibrium reconstruction code was 20 Pa. Therefore, it is inferred that the fast electron pressure is dominant. In order to obtain sufficient TS signal without such accumulation and to enhance the SN ratio, a multiple-pass TS measurement system with an optical cavity has been developed. In the cavity, a laser pulse can be confined coaxially between two mirrors by a fast switching of the voltage applied to the pockels cell. The first coaxial multiple-pass TS measurement for tokamak plasmas has been performed successfully, with a factor of 5 greater photon counts compared to the conventional single-pass configuration. A new optical system with improved optical efficiency is being developed. Results obtained with the improved system will be presented.

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Measurement of lower-hybrid waves with microwave back-scattering on TST-2

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Non-inductive plasma start-up with lower-hybrid (LH) waves is being investigated on the TST-2 spherical tokamak at the University of Tokyo. Plasma current ramp-up up to 20 kA has been achieved so far with a capacitively-coupled combline (CCC) antenna. The current drive efficiency dropped sharply around the mode conversion density limit that increases with the magnetic field. Full-wave modeling is underway to describe quantitatively the LH current drive on TST-2. Direct wave measurement in the plasma core is important for quantitatively accurate analysis. A new microwave back-scattering diagnostic is fabricated for this purpose. A microwave probe beam in the range of 10-40 GHz are launched into the plasma and scattered by the density fluctuations of the LH waves. The scattered light is detected by mixing it with the swept microwave source (swept frequency homodyne detection). Since the signal amplitude at the microwave frequency is proportional to the density fluctuation amplitude at the microwave wavenumber, the radial wavenumber spectrum of the LH waves can be measured. The actual wave measurements will be presented if available.

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Nonlinear Fishbone Dynamics in Spherical Tokamaks with Toroidal Rotation

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Abstract

Fishbone is ubiquitous in tokamak plasmas with fast ions. A numerical study of nonlinear dynamics of fishbone has been carried out in this work. Realistic parameters of NSTX are used to understand linear instability and nonlinear frequency chirping in real tokamak plasmas. First, the effects of shear toroidal rotation are considered for fishbone instability. It is shown that with low q_{min} , toroidal rotation has small effects on the mode; while with high q min , a new unstable region with a strong ballooning feature in mode structure appears. Second, a systematic study of nonlinear frequency chirping and energetic particles' dynamics is carried out. It is found that, linearly, the mode is driven by both trapped particles and passing particles, with resonance condition $\omega_d \simeq \omega$ for trapped particles and $\omega_\phi + \omega_\vartheta \simeq \omega$ for passing particles, where ω_d is trapped particles precession frequency, and ω_ϕ, ω_ϑ are passing particle transit frequency around toroidal and poloidal direction. As the mode grows, resonance particles oscillate and move outward in P_ϕ space, which reduces particles' frequency. We believe that this is the main reason for the mode frequency chirping down. Finally, as the mode frequency chirping down, particles with lower orbit frequencies, which are non-resonant linearly, can turn into resonant particles in the nonlinear regime. This effect can sustain a quasi-steady state mode amplitude observed in the simulation.

Distinct turbulence sources and confinement features in spherical tokamak plasma regime

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Highly distinct features of spherical tokamaks (ST) such as NSTX/U result in a different fusion plasma regime with unique physics properties compared to conventional tokamaks. Nonlinear global gyrokinetic simulations critical for addressing turbulence and transport physics in ST regime have led to new insights regarding non-traditional turbulence sources contributing to plasma transport and confinement in ST experiments. The drift wave Kelvin-Helmholtz (KH) instability characterized by intrinsic mode asymmetry is identified in strongly rotating NSTX L-mode plasmas. For the first time, the KH mode is shown as a driver of significant transport in realistic fusion experiments. Also for the first time, long wavelength, quasicoherent dissipative trapped electron modes are found to be excited over a wide range of NSTX parameter regime despite the presence of strong $\mathbf{E} \times \mathbf{B}$ shear, providing a robust turbulence source dominant over the traditional collisionless trapped electron modes in ST plasmas. Furthermore, DTEMdriven transport in NSTX parametric regime is shown to increase with electron collision frequency, offering one possible source for the confinement scaling observed in experiments. More interestingly, the existence of a minimum plasma transport regime that future advanced STs may access is predicted. This work was supported by U.S. DOE Contract DE-AC02-09CH11466.

The investigation of the colorimetry to measure the deposition thickness on the plasma-facing wall in QUEST

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After several campaigns in QUEST, the deposition has been formed on the plasma-facing wall (PFW), which is originally made of stainless steel. The main ingredient is the mixture of carbon and metal such as iron, nickel, chromium, and tungsten. Especially tungsten could come from the divertor and limiter composed of tungsten mono-blocks. The film thickness is the important parameter to understand hydrogen-isotope retention and its recycling property [1]. Its profile in the whole of the PFW should be estimated to understand hydrogen retention and particle balance. To measure the whole profile of the film thickness, an innovative method named colorimetry has been developed, however its application was limited in carbon dominant deposition film [2]. The colorimetry measures the reflectivity on the PFW, which is related with the film thickness and complex refractive index of deposition. Therefore, the complex refractive index of the deposition should be investigated to obtain the film thickness. In QUEST, the complex refractive index was estimated with plasma-exposed samples with the help of the TEM and ellipsometer, and it was quite different from carbon. The film thickness of the PFW measured with the colorimetry agreed with that of the plasma-exposed samples measured with the ellipsometer. This indicates the colorimetry is possible to be applied to the mixed-material deposition as formed in QUEST.

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Comparing the Magnetic Divertor Topology and Transport with Resonant Magnetic Perturbation Fields in High and Low Aspect Ratio Tokamaks

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Explorations are under way to optimize the magnetic divertor configuration of NSTX-U with the goal of improving neutral and impurity fueling and exhaust. The application of resonant magnetic perturbation (RMP) fields is being considered to spread heat and particle fluxes in the divertor, control impurity transport, and adjust the plasma refueling and neutral exhaust cycle. Given the number of next generation tokamak design studies using both high and low aspect ratio approaches, it is critical to understand the relationship between the perturbed magnetic configuration and edge plasma transport, and how this is affected by the aspect ratio. Similar investigations to those being completed for NSTX-U have already been carried out at the high aspect ratio DIII-D tokamak in San Diego. A comparison is made between the high-aspect ratio DIII-D tokamak (R/a~2.7) and the low aspect ratio NSTX-U tokamak (R/a~1.7), using the EMC3-EIRENE fluid plasma and kinetic neutral transport code. A standard poloidal divertor configuration with an n = 3 RMP field applied is used to compare the effects of RMP fields in these two devices with high and low aspect ratio, respectively. This work is funded in part by the Department of Energy under grant DE-SC0012315, DE-FC02-04ER54698 and by startup funds of the Department of Engineering Physics at the University of Wisconsin-Madison.

Investigation of electron energization mechanism during merging startups developed in UTST

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Electron energization mechanism during merging startup is experimentally investigated in the University of Tokyo Spherical Tokamak (UTST). Magnetic probe and Langmuir probe measurements indicate that the diffusion region extends significantly in the downstream direction and also that the size of the diffusion region is not affected by the toroidal magnetic field. Trajectory calculation using the measured profiles of magnetic field and electric field indicates that the electrons are energized by field-aligned electric field within the diffusion region. In the condition of same initial plasma current, the electron energy increases with the toroidal magnetic field.

An optimized merging startup method with longer startup phase is developed in UTST in order to enhance the plasma heating by magnetic reconnection. In the present setup, the plasmas are initiated at the null points transiently formed by the coil current and wall eddy current [1], therefore the formation time is as short as 50μsec. We will discuss the feature of the STs formed with these methods.

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Equilibrium Reconstruction of Detailed Current Density Profile Structure from External and Internal Magnetic Measurements in VEST

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A detailed current density profile structure is obtained in VEST by full equilibrium reconstruction from external and internal magnetic measurements. Vertical component of the magnetic field is measured by magnetic Hall sensors from inside the plasma, in addition to the typical external magnetic diagnostics. Grad-Shafranov equation solver TEFIT is modified to incorporate internal magnetic, external magnetic and diamagnetic measurements. The reconstructed equilibrium parameters are compared to the discharge characteristics of VEST to check the consistency of the result. It is shown that with the internal magnetic probe data, the uncertainty of the current density profile by equilibrium reconstruction is reduced. From the reconstructed current density data, the transition from hollow to peaked profile is observed, which is a typical phenomena during a current ramp up in a tokamak. The current penetration phenomena will be discussed in more detail with the established equilibrium reconstruction procedure including the internal magnetic probe data in addition to the typical external magnetic diagnostics.

Particle-loss Control for Making RF-induced Breakdown in QUEST

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RF-induced breakdown will be applied in ITER [1] and plays an essential role in plasma initiation in spherical tokamaks (STs). The process of breakdown in tokamaks is simply described as competition between plasma production and particle-loss [2], however, quantitative investigation still remains insufficient. In this study, the combinative investigation of RF-induced breakdown experiments with altering magnetic configuration and a point model of hydrogen ionization [3] was performed to determine typical duration for particle-loss (τ). Experiments have been performed on QUEST, which is a middle-sized ST equipped with electron cyclotron wave at frequency of 2.45 and 8.2 GHz. Magnetic configurations of positive and negative n-index at electron cyclotron resonance (ECR) layer can be controlled in combination with poloidal field coils. Experiments were carried out at both frequency of 2.45 GHz, 10kW and 8.2 GHz, 12kW in positive and negative n-index configurations.

To give quantitative evaluation to the breakdown procedure, it is necessary to specify τ experimentally and verify whether it is consistent with the model. The point model that developed takes into account possible reactions of hydrogen such as excitation, ionization and recombination. The initial values of electron temperature and electron density for this model were estimated by diverter probe measurement. The conventional particle-loss mechanism during breakdown phase was investigated by the point model with experiments of negative n-index configurations where electron loss parallel to magnetic line is dominantly contributing to τ. In QUEST, the estimated connection length ($\mathrm{L} = C_S \cdot \tau \sim 150\mathrm{m}$) as breakdown threshold for negative n-index experiment was consistent with L calculated from the model. With positive n-index configurations, breakdowns were easily obtained compared with negative n-index. Furthermore, the experimental results indicate that τ contains the terms of not only connection length but also electrons confined in magnetic structure. The current point model does not describe this behavior of confined electrons and it requires quantitative valuation of how much these electrons contribute to τ. Diagnostics by means of velocity space and trajectory calculation are beneficial for that purpose. Further information will be discussed in presentation.

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