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Book of Abstracts

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Overview

OV — Oral

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Overview of KSTAR Initial Experiments

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In 2009 campaign, main emphasis is put on the understanding of magnetics since the jacket of the cable-in-conduit cable (CICC) of the superconducting magnets are made of incoloy 908, which is a slightly magnetic material. A numerical model including the effect of magnetic materials has been developed and the result has been verified by measuring the effect of residual magnetic field by using electron beam and the Hall probe array system. In this campaign, the machine was operated with the toroidal magnetic field of up to 3 T. Circular plasmas with current of 300 kA and pulse length of 2 s, have been achieved with limited capacity of PF magnet power supplies. The second harmonic pre-ionization with 110 GHz, 250 kW gyrotron at 2 T has been studied. Various parameters such as injection angle, position and pressure have been scanned to optimize the pre-ionization. The ICRF Wall Conditioning (ICWC) was routinely applied during the shot to shot interval. The effect of ICWC has been quantitatively assessed by dedicated diagnostic systems. For the study of in-vessel dust characterization, duct collectors have been installed and coupons have been installed to study the campaign-integrated deposition characteristics.

Since the successful achievement of the first plasma in 2008, KSTAR has been upgrading its performances. Since the operation plan in 2010 includes operations with D-shaped and diverted plasmas, all of the plasma facing components (PFCs) and divertor will be installed in the vacuum vessel. Although the divertor will be eventually covered with carbon fiber composite (CFC) for long pulse operation, all the PFCs and divertor will be covered with actively cooled graphite tile for short pulse until 2012. The sixteen-segmented in-vessel control coils (IVCCs) will also be installed prior to the installation of the PFCs. The IVCCs will be externally connected to form two sets of circular coils for vertical and radial position control, and then the IVCCs will be additionally connected to form twelve picture-framed coils for RWM/FEC control later. The first NBI system (NBI-1), designed to deliver 8 MW D0 neutral beams into the KSTAR plasmas with three ion sources, is now under fabrication and will be commissioned for the 1 MW beam power with the first ion source in 2010.

OV/1-2

Recent Progress in High Power Heating and Long Pulse Experiments on EAST

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New capabilities including RF powers and diagnostics have been developed since last IAEA meeting. Totally about 3 MW H&CD power is injected into the EAST plasma. Behavior of basic plasma confinement/transport under the condition of dominant electron heating and low momentum input is investigated. Divertor performance has been systematically assessed for both SN and DN configurations with normal and reversed B_t directions. Intriguing findings, particularly, in asymmetries correlated with classic drift and in control by gas puffing at various divertor locations are studied and comparison with SOLPS modeling is being done. 1 MW LHW power injection has produced fully non-inductive plasma discharges. Such discharges have been readily extended well over one minute. Higher power up to 1.5 MW led over current driven. LHCD results fitted to the K-F model verifies classic collision picture of wave energy converted to poloidal magnetic energy, which is further confirmed by the simulations. Both direct electron and ion heating by ICRF has been observed in MC scenario, implying direct wave (ICW via MC) damping on ion. Effort is being dedicated to different heating scenarios of ICRH to address several key physical issues, such as transport control, flow driven, etc. Novel wall conditioning techniques using RF and HF glow discharges in the presence of toroidal magnetic fields have been successfully applied as well as real-time coating during plasma discharges to achieve a good wall condition. The effects of these techniques on the first wall and plasma performance are analyzed and assessed for ITER. Great efforts are being dedicated to integration of ICRH and LHCD scenarios, divertor configuration control coupled with optimized puffing and active pumping, under well conditioned wall to achieve improved plasma performance. Comparisons are being made with integrated modeling and data analysis based on IMFIT and PTRANSP.

OV/1-3

Overview of JET Results

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Since the last IAEA Conference JET has been in operation for one year with a programmatic focus on the qualification of ITER operating scenarios, the consolidation of ITER design choices and preparation for plasma operation with the ITER-like wall presently being installed. Good progress has been achieved, including stationary ELMy H-mode operation at 4.5 MA. In matched NBI+ICRF heated H-mode plasmas with up to 85% of ICRF heating the confinement was found to be independent of the heating mix. The high confinement hybrid scenario has been extended to high triangularity, lower ρ^* and to pulse lengths comparable to the resistive time. The steady-state scenario has also been extended to lower ρ^* and ν^* and optimised to simultaneously achieve, in stationary conditions, ITER-like values of all other relevant normalised parameters. Dedicated physics studies have demonstrated reduced core ion stiffness in plasmas with high rotational, and low magnetic, shear, allowing gradients with normalised ion temperature gradient lengths up to 8 in the fastest rotating hybrid discharges. Effective sawtooth control by fast ions has been demonstrated with ^{3}He minority ICRH, a scenario with insignificant minority current drive. This is encouraging for ITER where magnetic shear modification by ICCD is expected to be difficult. ELM control studies using external $n = 1$ and $n = 2$ perturbation fields have found a resonance effect in ELM frequency for specific q_{95} values. Complete ELM suppression has however not been observed, even with an edge Chirikov parameter larger than 1. For pellet pacing it has been found that to reliably trigger an ELM the pellet needs to be sufficiently large (and fast) to penetrate to the top of the pedestal. In disruption studies with Massive Gas Injection up to 50% of the thermal energy could be radiated before, and 20% during, the thermal quench. Halo currents could be reduced by 60% and, using argon/deuterium and neon/deuterium gas mixtures, runaway electron generation could be avoided. Most objectives of the ITER-like ICRH antenna have been demonstrated; matching with closely packed straps, ELM resilience, Scattering Matrix Arc Detection and operation at high power density (6.2 MW/m^2) and antenna strap voltage ($>$ 40 kV). Coupling measurements are in very good agreement with TOPICA modelling.

Work supported by EURATOM and carried out under EFDA.

OV/1-4

DIII-D Contributions toward the Scientific Basis for Sustained Burning Plasmas

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DIII-D is making significant contributions to a scientific basis for burning plasma operation in ITER and future devices. These include explorations of increasingly reactor relevant scenarios, studies of key issues for projecting performance, development of techniques for handling heat and particle efflux, and assessment of key issues for the ITER Research Plan. The four main ITER operating scenarios are a major focus. The baseline H-mode exhibits a $\sim 15\%$ confinement decrease at low rotation, but still projects to $Q = 10$ in ITER. These experiments included a first demonstration of preemptive neoclassical tearing mode suppression in this scenario. Joint DIII-D/JET ρ^* scans in the hybrid regime imply Bohm-like confinement scaling. A variety of q profiles were evaluated to optimize high performance steady state operation. Startup and shutdown techniques were developed for the restrictive environment of future devices while retaining compatibility with advanced scenarios. With its flexible heating and current drive systems and broad diagnostic set, DIII-D provides key tests of theory based physics models. Detailed sets of turbulence measurements are being obtained for comparison with GYRO. Studies of intrinsic rotation and non-resonant magnetic fields (NRMF), which “drag” the plasma to a non-zero velocity, increase our understanding of rotation. Joint DIII-D/JET experiments indicate a weak, inverse dependence of the pedestal width on ρ^*. DIII-D research addresses methods to control particle and energy flows leaving the plasma. Recently, resonant magnetic perturbation (RMP) edge localized mode (ELM) suppression was demonstrated with a radiative divertor. ELM-free QH-mode operation has been extended to low applied torque via application of NRMF. Massive gas injection and large shattered pellets have demonstrated delivery of particles for safe shutdown to mitigate disruption. The application of an RMP field to deconfine runaway electrons can reduce the delivery system requirements. DIII-D addresses specific issues for ITER. The L-H threshold increases from D to He $(30-50\%)$ to H $(\times 2$ above D). Experiments carried out with a mockup of one of ITER’s TBMs indicate that no serious effects are anticipated. Detailed analysis will aid in validating physics models for the effects of 3D fields.

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OV/2-1

Progress toward Ignition on the National Ignition Facility

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The National Ignition Facility (NIF) at Lawrence Livermore National Laboratory was formally dedicated in May 2009. The hohlraum energetics campaign with all 192 beams began shortly thereafter and ran until early December 2009. These experiments explored hohlraum-operating regimes in preparation for ignition experiments with layered cryogenic targets in late FY2010. The series culminated with an experiment that irradiated an ignition scale hohlraum with 1.05 MJ, approximately thirty times the energy that any hohlraum has been irradiated previously. The results demonstrated the ability to produce a 285 eV radiation environment in an ignition scale hohlraum while meeting ignition requirements for symmetry, backscatter and preheat. Complementary scaling experiments indicate that with 1.3 MJ, we will increase the capsule drive temperature to more than 300 eV, the point design temperature for the first cryo layered implosion experiments. Preparation for these implosions included installation of a variety of nuclear diagnostics, cryogenic layering target positioner, advanced optics, and facility modifications needed for tritium layered targets and for routine operation at laser energy greater than 1.3 MJ.

This talk presents the findings and conclusions from the ignition hohlraum energetics experiments carried out in 2009. It also provides an update on the initial cryo layered implosions and the companion experiments to optimize the compressed fuel assembly.

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OV/2-2

Progress in ITER Construction

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Since the entry into force of the ITER Agreement in October 2007, the ITER Organization and the Domestic Agencies have made significant progress in the finalization of the ITER design and the establishment of the baseline. This paper will describe the main advances and achievements since the last IAEA conference.

The Integrated Project Schedule has gone through a number of iterations following interactions with Domestic Agencies and their industries. With operating a First Plasma at the end of 2019 construction will be finished and the functionality of all major components will be demonstrated. Through a series of operation and installation campaigns in the following seven years full DT operation should be reached by 2026. The associated resource estimate for the cost of the ITER Organization has been reviewed by an independent panel and has been developed using common project management practices. Attention has been paid to the development of risk based assessments of the schedule giving confidence that this schedule can be achieved if no major unforeseen events happen.

The ITER platform has been prepared by Agence ITER France and was finalized by the summer of 2009. The next stage will be the start of the excavation and construction of the anti-seismic support structure for the tokamak building. The road between the harbor of Marseille and the Cadarache site is ready to receive the first test convoys. Just under 90% of the components for ITER will provided in-kind by the Members through so-called Procurement Arrangements, which are bi-lateral agreements between the ITER Organization and the Domestic Agencies. Until now 34 Procurement Arrangements have been signed, representing almost half of the value of the total in-kind contributions. The first Procurement Arrangements concentrated on the long-lead items, such as the Magnet systems, Buildings and the Vacuum Vessel. Most of the Procurement Arrangements for these long-lead items have now been transformed into procurement and production contracts by the Domestic Agencies.

The other major component, the Vacuum Vessel, has a final design with all In-Vessel coil interfaces integrated and a contract in Korea has been placed for two sectors while the call for tender for the other seven sectors is underway in Europe.

OV/2-3

Overview of JT-60U Results toward the Resolution of Key Physics and Engineering Issues in ITER and JT-60SA

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This paper overviews recent results in JT-60U, which cover urgent issues in ITER and JT-60SA. Highlighted topics are (i) rotation effect on ELM stability and impurity transport, (ii) ELM control by pellet and edge ECH and (iii) interaction between RWM and ELM. Detailed investigation on momentum transport in a plasma with ITB has shown that amplitude of velocity perturbation propagating from the outside of ITB significantly reduces at the ITB, and that momentum transport is improved inside the ITB. Numerical analysis on the mechanism of the destabilization of the edge-localized MHD modes by sheared toroidal rotation using the MINERVA code has newly revealed that the edge-localized MHD mode is destabilized by the difference between the eigenmode frequency and the equilibrium toroidal rotation frequency, and that the destabilizing effect is stronger for high toroidal mode numbers. Statistical analysis of the growth rate of type-I ELM precursor has clarified that the growth rate is 10^{-3} of the Alfven frequency, suggesting that the instability with such a small growth rate determines the onset condition of type-I ELM. Integrated simulation of pellet-triggered ELM showed two mechanisms for ELM onset: (i) radial redistribution of background plasma pressure through the energy absorption by the pellet and subsequent penetration due to the $\mathbf{E} \times \mathbf{B}$ drift and (ii) transient enhancement of heat and particle transport caused by the sharp increase in local density and temperature gradients in the vicinity of ablated cloud. Energetic particle driven Wall Modes

(EWMs), which are destabilized in a high-beta regime above the no-wall limit, have been found to trigger ELM and decrease the ELM amplitude by about 1/3 with increasing ELM frequency. A new inward pinch effect of high-Z impurities due to the change in the drift velocity attributed to the change in the charge state on the drift orbit in toroidally rotating tokamak plasmas has been discovered and confirmed by analytic calculation and IMPGYRO Monte Carlo simulation. Furthermore, new records on (i) pulse duration and output power of ECRF (1.5 MW for 4 s and 1 MW for 17 s) and (ii) beam energy and beam current of NNB (507 keV at 1 A and 486 keV at 2.8 A) with voltage-holding capability of about 200 kV per single stage have been achieved, which ensure and extend the heating and current drive capability of ITER and JT-60SA.

OV/2-4

Overview of Physics Results from NSTX

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During the last two experimental campaigns, the low aspect ratio NSTX has explored physics issues critical to both toroidal confinement physics and ITER. Experiments have made extensive use of both lithium coatings for wall conditioning and external non-axisymmetric field correction to reliably produce high-performance discharges with non-inductive current fractions of up to 70, extending to 1.7 s in duration. The error-field correction coils have been used to trigger ELMs for controlled ELM pace-making with high reliability and have also contributed to an improved understanding of both neoclassical tearing mode and resistive wall mode (RWM) physics. RWM research has pointed to the interplay between rotation and kinetic effects for mode stability. High Harmonic Fast Wave (HHFW) heating produced plasmas with $T_e(0)$ in excess of 6 keV, and was used as a tool to study the species dependence of the L-H transition, which indicated that the power threshold for D and He could be comparable, suggesting that operation in helium may be the best approach to developing H-mode scenarios in the early non-nuclear phase of ITER operation. Recent results from a Fast Ion D_α diagnostic show a depletion of the fast ion profile over a broad spatial region as a result of toroidicity-induced Alfvén eigenmodes (TAE) and energetic particle modes (EPM) bursts. In addition, it is observed that other modes (e.g. Global Alfvén eigenmodes) can trigger TAE and EPM bursts, suggesting redistribution of fast ions by high-frequency AEs. NSTX results also show the pinch velocity to decrease as the collisionality is reduced a result of particular importance to ITER as it will have limited external momentum input. In support of tritium retention studies in ITER, the processes governing deuterium retention by graphite and lithium-coated graphite plasma facing components (PFCs) was investigated. A novel "snowflake" divertor configuration, was tested and many beneficial aspects were found. A reduction in the required central solenoid flux has been realized when discharges initiated by coaxial helicity injection were ramped in current using induction. Other experiments have been conducted to address research of high priority to the ITPA and ITER.

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OV/2-5

Overview of Results from the Large Helical Device

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Physical understanding of net-current free helical plasmas has progressed remarkably in the Large Helical Device (LHD) since the last Fusion Energy Conference in Geneva, 2008. Highlighted results from the LHD experiments in these two years are overviewed. In parallel with parameter improvement, important physics processes of transport and MHD for fusion energy development have been identified and assessed by the analysis based on profile diagnostics with fine spatial resolution and numerical computation to cope with a real 3D geometry. LHD is based on the heliotron employing a pair of superconducting helical coils. The primary heating source is NBI with a heating power of 23 MW, and ECH with 3.5 MW plays an important role in local heating and power modulation in transport studies. The maximum central density has exceeded 1×10^{21} m^{-3} due to the formation of the Internal Diffusion Barrier (IDB) at the magnetic field of 2.5 T. A Resonant Magnetic Perturbation (RMP) with $m/n = 1/1$, which has resonance in the plasma periphery, has demonstrated the radial expansion of a super-dense-core surrounded by an IDB through the density reduction in the mantle outside the IDB. The plasma with a central ion temperature reaching 5.6 keV exhibits the formation of an Internal Transport Barrier (ITB). The ion thermal diffusivity decreases to the level predicted by neoclassical transport. This ITB is accompanied by spontaneous toroidal rotation and an Impurity Hole which generates an impurity-free core. This phenomenon is due to a large outward convection of carbon impurities in spite of the negative radial electric field. The magnitude of the Impurity Hole is enhanced in the magnetic configuration with larger helical ripple and for higher Z impurities. Another mechanism to suppress inpurity contamination is impurity screening at the plasma edge with a stochastic magnetic field. The operational envelope of high beta has been extended to 5.1%. The demand for 3D modeling is becoming inevitable for accurate and detailed studies in tokamaks. A helical system shares common 3D related physics issues with tokamak such as documentation of 3D equilibria, transport in a stochastic magnetic field, plasma response to RMP and divertor physics. Results from LHD would accelerate tokamak research in addition to advancing the prospect for a helical reactor.

OV/3-1

Overview of ASDEX Upgrade Results

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The ASDEX Upgrade programme is directed towards physics input to critical elements of the ITER design and the preparation of ITER operation, as well as addressing physics issues foreseen for a future DEMO design. After the finalization of the tungsten coating of

the plasma facing components, re-availability of all flywheel-generators allowed high-power operation with up to 20 MW heating power. Implementation of alternative ECRH schemes (140 GHz O_2^- and 10^5 GHz X3-mode) facilitated central heating above $n_e = 1.2 \times 10^{20}$ m^{-3} and low q_{95} operation at $B_t = 1.7$ T. ECRH O_2-heating was used to maintain low central tungsten concentrations in high P/R discharges with 20 MW total heating power and divertor load control with nitrogen seeding. Improved energy confinement is obtained with nitrogen seeding both for type-I and type III ELMy conditions. The main contributor are increased plasma temperatures both in the core and at the pedestal, no changes of the density profile have been observed. Fast ion losses induced by shear Alfven eigenmodes have been investigated by time-resolved energy and pitch angle measurements. While single Alfven cascades (AC) and toroidal Alfven eigenmodes (TAE) eject resonant fast ions in a convective process, the overlapping of AC and TAE spatial structures was found to lead to a large fast ion diffusion and loss. Doppler reflectometry studies at the L-H transition allowed the disentanglement of the interplay between the oscillatory geodesic acoustic modes (GAMs), turbulent fluctuations and the mean equilibrium $\mathbf{E} \times \mathbf{B}$ flow in the edge negative E_r well region just inside the separatrix. The transition of the GAM into low frequency fluctuations is a candidate for the long sought-after trigger for the $\mathbf{E} \times \mathbf{B}$ velocity feedback loop believed to be responsible for the L-H transition. Further improvements of the pedestal diagnostics including the application of integrated data analysis techniques revealed a refined picture of density and temperature profile evolution during an ELM cycle. Different time scales for the recovery of T_e and n_e gradients after an ELM crash were identified, finally the gradients evolve into a highly fluctuating phase. Currently, the first set of in-vessel coils for ELM control is being installed in ASDEX Upgrade, and first results will be reported if available.

OV/3-2

Overview of Recent Results from Alcator C-Mod including Applications to ITER Scenarios

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Alcator C-Mod combines high magnetic field, advanced shaping and divertor configurations, and the ability to operate with solid all-metal plasma-facing components. C-Mod accesses regimes of extreme edge power density with SOL power widths of order of a few mm implying mid-plane parallel power flows > 1 GW/m^2, surpassing the design for ITER. We have significantly extended the I-mode regime to high power and plasma performance. I-mode yields strong edge ion and electron temperature barriers, excellent energy confinement ($H_{ITER-98}$ up to 1.2), and low collisionality. The I-mode regime has no need for ELMs to maintain density and impurity control. Experiments to simulate ITER-like plasma evolution during startup and rampdown have been carried out on C-Mod using the ITER shape and magnetic field, with comparable safety factor, normalized pressure and energy confinement scaling. During ramp-up, with early divert times and transition to H-mode, significant loop-voltage savings are realized, as predicted for ITER from TSC simulations. Detailed studies of ICRF-induced flow drive on C-Mod reveal that the effi-

ciency depends strongly on He^3 concentration in the $D(He^3)$ mode conversion regime, with driven core toroidal rotation up to 110 km/s (M $\sim$ 0.3). Experimental and theoretical studies of intrinsic rotation show that central toroidal rotation, observed in the absence of external momentum input, scales with edge temperature gradient, and the relationship to fluctuation-induced residual stress is under investigation. For $n_e > 1 \times 10^{20}$ m^{-3} LHCD efficiency drops off more rapidly than expected theoretically, and mechanisms of anomalous absorption in the edge plasma are under investigation. Results from a new, advanced Lower Hybrid launcher, aimed at low-loss and high power density ($\sim$ 100 MW/m^2) will be reported. Lower Hybrid waves have been used to produce a seed population of non-thermal electrons ($E >$ 100 keV), which can be accelerated during the thermal quench (TQ) phase of disruptions up to $\sim$ 20 MeV. Modeling using the 3D NIMROD code shows that, in these conditions, when massive gas puffing is applied for disruption mitigation, the strong MHD activity which grows during the TQ causes a nearly complete stochasticization of the magnetic field, in turn causing loss of the runaway electrons during the TQ.

OV/3-3

Overview of Physics Results from MAST

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Important advances have been made on MAST, aided by substantial technical developments including internal ELM control coils, a disruption mitigation valve (on loan from FZ Jülich), a divertor science facility and major diagnostic enhancements e.g. multi-chord MSE, Thomson Scattering (TS) upgrade (part funded by University of York) etc. To study the physical mechanisms underlying the L-H transition, evolution of temperature, density and radial electric field profiles has been measured with very high spatial and temporal resolution and the edge current density profile has been determined from MSE measurements. The H-mode power threshold is reduced if the distance between the X-point and the strike point is shortened and it is 50% higher for helium discharges (with D beams) than for deuterium. MAST H-modes exhibit a weaker q and stronger collisionality dependence of heat diffusivity than implied by IPB98$_{(y,2)}$ scaling. Equilibrium flow shear has been included in gyro-kinetic codes, improving comparisons with results from strongly rotating MAST plasmas. Poloidal rotation is measured to be low and hence makes only a small contribution to the net $\mathbf{E} \times \mathbf{B}$ flow shear in MAST. The upgraded TS system is also being used to study pellet ablation and post-pellet transport. ELM mitigation by resonant magnetic perturbations (RMPs) has been demonstrated using both internal and external coils but full stabilization has not been observed. Non-linear MHD modelling indicates that strong rotational screening may inhibit RMP penetration in MAST H-mode plasmas with Type I ELMs. Retarding Field Analyzer measurements show high ion temperatures in the scrape-off layer of L-mode plasmas and, in the presence of Type I ELMs, ions with energy greater than 500 eV are detected 20 cm outside the separatrix. Disruption mitigation by massive gas injection reduced divertor heat loads by up to 70%. MAST plasmas with q greater than 1 and weak central magnetic shear regularly exhibit a

long-lived saturated ideal internal mode. Measured plasma braking in the presence of this mode compares well with neoclassical toroidal viscosity theory. The upgraded TS system has enabled detailed analysis of the density and temperature profiles in and around a neoclassical tearing mode island, permitting tests of models for the critical island width.

Work supported by UK EPSRC and EURATOM.

OV/3-4

Contribution of Tore Supra in Preparation of ITER

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Tore Supra routinely addresses the physics and technology of very long duration plasma discharges, thus bringing vital information on critical issues of long pulse/steady-state operation of ITER.

A new ITER relevant LHCD launcher, based on the PAM concept, has allowed coupling to the plasma a power level of 2.7 MW for 35 s, corresponding to a power density close to the design value foreseen for an ITER LHCD system. In accordance with the expectations, excellent long distance (10 cm) power coupling has been obtained.

Successive stationary states of the plasma current profile have been controlled in real time featuring i) control of the presence/absence of sawteeth with varying plasma parameters, ii) obtaining and sustaining a “hot core” plasma regime without MHD activity, iii) recovery from a voluntarily triggered deleterious MHD regime.

A significant part of the Tore Supra programme has been devoted to experiments in direct support of ITER operation. The SOL parameters and power deposition have been documented during L-mode ramp-up phase, a crucial point for ITER before the X-point formation. Disruption mitigation studies have been conducted with massive gas injection, evidencing the difference between He and Ar and the possible role of the $q = 2$ surface in limiting the gas penetration. Finally, ICRF assisted wall conditioning in the presence of magnetic field has been investigated, culminating in the demonstration that this conditioning scheme allows to recover normal operation after disruptions.

An extensive characterisation of micro-turbulence performed during dedicated dimensionless scans on Tore Supra has been conducted. The impact of changing dimensionless parameters on wavenumber spectrum has been investigated, suggesting a transition in the turbulence regime when varying the collisionality. Turbulence measurements have also allowed to quantitatively compare experimental results to predictions by the 5D gyrokinetic code GYSELA. Numerical results simultaneously match the magnitude of effective heat diffusivity, rms values of density fluctuations, and wave-number spectra. Dynamics of edge turbulent fluctuations has been studied by correlating data from fast imaging cameras, reciprocating Langmuir probes and Doppler reflectometry. Filaments, localized on the LFS, were observed with qualitatively similar dynamics to the one observed in X-point devices.

OV/4-1

From Fast Ignition Realization Experiments (FIREX) to Electric Power Generation (LIFT)

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After a 50 years journey since the innovation of lasers, controlled ignition and subsequent burn will be demonstrated within a couple of years at the US National Ignition Facility (NIF). Fast ignition has the high potential to ignite a fuel using only about one tenth of laser energy of NIF [1]. This compactness may largely accelerate inertial fusion energy development. One of the most advanced fast ignition programs is the Fast Ignition Realization Experiment (FIREX) [2]. The goal of its first phase is to demonstrate ignition temperature of 5 keV, followed by the second phase to demonstrate the ignition-and-burn. The first experiment of FIREX-I gives a confidence that one can achieve ignition temperature at the 10 kJ heating laser energy. Given the demonstrations of the ignition temperature at FIREX-I and the ignition-and-burn at NIF, the inertial fusion research would then shift from the plasma physics era to electric power era.

[1] S. Atzeni, Phys. Plasmas 6 (1999) 3316.

[2] H. Azechi et al., Nucl. Fusion 49 (2009) 104024.

OV/4-2

Overview of FTU Results

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Since last IAEA-FEC, FTU has extended operations with Liquid Lithium Limiter (LLL) to assess its potentials as a plasma facing component in harsh fusion plasmas environment. The use of lithized walls had widened FTU parameter space producing ohmic plasmas at peaked density profile (peaking factor 2.5) up to 1.6 times Greenwald limit. New discharges at 700 kA confirm the spontaneous transition to an enhanced confinement regime, $1.2-1.3$ times ITER-97-L, for density values characteristic of the saturated ohmic confinement regime, when density peaking factor is above a threshold value of $1.7-1.8$. The ameliorated confinement is ascribed to the stabilization of ITG modes. D_2 pellets have been injected for the first time in Lithized discharges. A preliminary analysis of particle transport shows a Bohm gyro-Bohm type diffusion coefficient.

RF heated plasma profited of cleaner plasma conditions. Lower Hybrid waves penetration was clearly demonstrated at, and above, ITER densities with good control of edge parameters obtained by using the external poloidal limiter and the lithized walls to control the recycling. Flexibility and reliability of Electron Cyclotron heating system allowed FTU contributing to ITER relevant issues such as MHD control: sawtooth crash was actively controlled and disruptions at density limit were avoided. ECRH injected perpendicularly, demonstrated to be more efficient than oblique's, reducing by a factor 3 the minimum electric field required at breakdown. The theoretical effort in developing model to interpret high frequency fishbones on FTU and other experiments continued. The GFLDR framework was used for implementing an extended version of the hybrid MHD gyrokinetic code HMGC. The upgraded version of HMGC will be able to handle fully compressible nonlinear gyrokinetic equations and 3D MHD.

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Effects of 3D Magnetic Perturbations on Toroidal Plasmas

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To lowest order tokamaks are two-dimensional (2D) axisymmetric magnetic systems. But small 3D magnetic perturbations (both externally applied and from plasma instabilities)

have many interesting and useful effects on tokamak (and quasi-symmetric stellarator) plasmas. Plasma transport equations that include these effects, especially on diamagnetic-level toroidal plasma rotation, have recently been developed [1]. The 3D magnetic perturbations and their plasma effects can be classified according to their toroidal mode number n: low n (1 to 5) resonant ($q = m/n$ in plasma) and non-resonant fields, medium n (due to toroidal field ripple), and high n (due to microturbulence). This paper concentrates on low and medium n perturbations. Low n non-resonant magnetic fields induce a neoclassical toroidal viscosity (NTV) that damps toroidal plasma rotation throughout the plasma toward an offset flow in the counter-I_p direction; recent tokamak experiments have confirmed and exploited these predictions by applying external low n non-resonant magnetic perturbations. Medium n perturbations have similar effects plus possible ripple trapping and resultant edge ion losses. A low n resonant magnetic field induces a toroidal plasma torque in the vicinity of the rational surface; when large enough it can stop plasma rotation there and lead to a locked mode, which often causes a plasma disruption. Externally applied 3D magnetic perturbations usually have many components; in the plasma their lowest n components are amplified by plasma responses, particularly at high beta. Low n plasma instabilities (e.g., NTMs, RWMs) cause additional 3D magnetic perturbations in tokamak plasmas; tearing modes can bifurcate the topology and form magnetic islands. Finally, multiple resonant magnetic perturbations (RMPs) can cause local magnetic stochasticity and influence H-mode edge pedestal transport. These various effects of 3D magnetic perturbations can be used to control the toroidal plasma rotation and possibly plasma transport in localized resonant regions of the plasma, e.g., for ELM control. The present understanding and modeling of these various effects, and key open issues for development of a predictive capability of them for ITER will be discussed.

[1] J.D. Callen, A.J. Cole and C.C. Hegna, Nucl. Fusion 49, 085021 (2009); Phys. Plasmas 16, 082504 (2009); Phys. Plasmas 17, 056113 (2010).

OV/4-4

Overview of TJ-II Experiments

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This paper presents an overview of experimental results on particle control using Li-coating, transport and plasma bifurcations in TJ-II. The results show a great improvement of plasma particle control after Li-coating, in comparison with the operation under Boron coated walls. The beneficial Li properties have a strong effect on this device allowing the access to improved confinement regimes. The effect of low order magnetic resonances have been also explored showing a local decreasing of heat diffusivity, while an increase of fast ion flux due to Alfven modes in NBI regime.

Perturbative experiments of gas injection have shown a transition from peaked (central impurity accumulation) to a broader profiles (with lower-central Z_{eff} profile). These profiles, tagged as "bell" and dome", play a role for the density limit achieved under pure NBI heating. Laser blow-off has been used to inject trace boron impurities to study impurity transport. The parallel transport of boron ions has been measured with two filter-copes monitoring B-II line on opposite sides of the torus.

The L-H transition appears during NBI injection and is characterised by the increase of diamagnetic energy, the decrease of H_α emission, the reduction of turbulence, and the development of steep density gradients. The turbulence reduction precedes the increase in the mean sheared flow, but is simultaneous with the increase in the low frequency oscillating sheared flow. This type of spontaneous transitions is added to the ones that happen at lower densities, which correspond to the shear flow development and can be also provoked by biasing. During both low and high plasma density bifurcations, the correlation length of the plasma potential becomes of the order of the machine size. The radial structure of plasma fluctuations has been investigated in the plasma boundary region of the TJ-II stellarator. Potential transport events are observed to be born in the proximity of the edge velocity shear layer and propagate outwards in the SOL, whereas they propagate inwards in the edge region, with radial velocities of $1 - 10$ km/s.

Electron Bernstein Waves heating experiments are carried using the OXB mode conversion. Those waves are also very efficient to drive non-inductive currents and hence can be used to tailor the rotational transform profile and to compensate bootstrap current.

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Overview of Experimental Results on the HL-2A Tokamak

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Since last FEC, recent experimental campaigns on HL-2A have been focused on studying and understanding the physics of H-mode, zonal flow and turbulence, transport, energetic particle, MHD activities and fueling. The ELMy H-mode operation has been achieved on HL-2A with ECRH and NBI. Both low frequency zonal flow and geodesic acoustic mode have been observed coexistence. Their dependence on safety factor and ECRH power is studied. The 3D spatial structures of blobs are investigated with the novel combination of a poloidal and a radial Langmuir probe arrays separated 210 cm toroidally. Especially, the large scale structure of a blob along a magnetic field line is first observed. Density modulation experiments with ECRH have been performed to determine separately particle diffusivity and convective velocity. New features of electron fishbone, including frequency upward and downward chirping, V-shape sweeping and frequency jumps, have been observed when $P_{ECRH} > 0.7$ MW.

OV/5-1

Studying Ignition Schemes on European Laser Facilities

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First experiments on the National Ignition Facility in the United States have successfully begun in 2009 which give confidence in demonstrating ignition and net energy gain within the next two years. Such an achievement - likely to be confirmed a few years later on the Laser MégaJoule in France - will be a major step towards determining the feasibility of Inertial Confinement Fusion. The current status of the LMJ programme, from the laser facility construction to the indirectly-driven central ignition target design, will be first presented, as well as experimental campaigns, conducted on the Ligne d'Intégration Laser and on LULI2000, as part of this programme. Special attention will be paid to experiments aiming at characterizing high-energy laser interaction with under-dense plasmas and to supporting theoretical analysis and simulations.

The viability of the Inertial Fusion Energy (IFE) approach strongly depends on our ability to address the salient questions related to efficiency of the target design and laser driver performances. In the overall framework of the European HiPER project, two alternative schemes both relying on decoupling target compression and fuel heating – fast ignition (FI) and shock ignition (SI) – are currently considered. After a brief presentation of the HiPER project's objectives, FI and SI target designs will be discussed. Theoretical analysis, 2D radiation-hydrodynamics, hybrid and PIC simulations, performed at the Polytechnic University of Madrid (Spain) and the University Bordeaux 1 (France) will help understanding the key unresolved issues of the two schemes. The on-going European experimental effort to demonstrate the viability of the two schemes on various laser facilities will finally be described. Two FI-relevant problems will be tackled: (i) influence of plasma density scale-lengths on the laser-to-electron energy transfer and (ii) fast electron beam transport in pre-compressed matter. Preliminary experimental results on shock ignition will also be presented.

These studies have been partly funded through the EURATOM IFE KiT programme, the HiPER Preparatory Phase and the LASERLAB-Europe II transnational access programme.

OV/5-2

Progress and Scientific Results in the TCV Tokamak

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The TCV tokamak has the dual mission of supporting ITER and exploring alternative paths to a fusion reactor. Its most unique tools are a 4.5 MW ECRH system with 7 real-time controllable launchers and a plasma control system with 16 independent shaping coils. Additional ECRH power, ion heating sources, and in-vessel MHD control coils are now envisaged. Recent upgrades in temperature, density, and rotation diagnostics are being followed by new turbulence and suprathermal electron diagnostics, and a new digital real-time network is being commissioned. The shape control flexibility of TCV has enabled the generation and control of the first "snowflake" X-point, in which both the poloidal field and its gradient vanish. The predicted increases in flux expansion and edge magnetic shear have been verified experimentally, and studies of EC-heated snowflake H-modes are ongoing. In conventional H-modes we have characterized the pedestal response to varying input power, showing that high power can open access to the second stable ballooning regime. In-out divertor asymmetries in ELM power deposition are studied with two fast infrared cameras. L-mode impurity transport increases strongly with triangularity, similar to energy transport. The relation between impurity and electron density gradients is explained in terms of neoclassical and turbulent drives. ECCD modulation techniques have been used to study the role of the current profile in energy transport, and simulations reproduce the results robustly. Studies of torqueless rotation have yielded a quantitative measurement of the rapid deceleration during the sawtooth crash and of the ensuing relaxation. A newly predicted mechanism for turbulent momentum transport associated with up-down plasma asymmetry has been verified in TCV. The role of MHD in rotation generation is currently studied, in parallel with the use of ECRH to affect tearing mode stability. Trapped electron mode turbulence measured by correlation ECE increases with increasing power and triangularity and with decreasing collisionality, consistent with the concomitant increase in anomalous transport and in qualitative agreement with quasilinear mixing-length estimates. Sawtooth period control and soft X-ray emission profile control have been demonstrated in TCV using the new digital control hardware, as a step on the way to more complex applications.

OV/5-3Ra

Overview of the RFX Fusion Science Program

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RFX-mod is a flexible 2 MA RFP ($R = 2$ m, $a = 0.46$ m), which is rapidly advancing its performance, to (a) explore the RFP approach to fusion; (b) provide state-of-the-art contribution to stability feedback control in fusion plasmas; (c) focus on the key topic of three-dimensional magnetic shaping in a growing partnership with the stellarator.

Self-organized single helical axis (SHAx) equilibria, with reduced magnetic fluctuations and strong electron internal transport barriers are documented [1]. Core T_e reaches 1.5 keV. Electron ITBs are located in a region of low/null magnetic shear, with hints of a peak in helical flow shear near q_{max}. Codes (GS2 and TRB) and theory indicate that in SHAx microtearings may drive transport across ITB, rather than ITG as in tokamak. Helical ITBs appear at Greenwald fraction ~ 0.2. To avoid this operational limit due to edge density accumulation, and to increase core density, a combination of H-pellet fuelling and lithization is explored. First lithization experiments result in better density control and more peaked density profiles. Significant increase of core density is obtained with H pellets.

Particle transport in the helical core is consistent with neoclassical and ORBIT shows that the $1/\nu$ regime (superbananas effects), typical of unoptimized stellarator, might not be present even at very low collisionality.

Codes originally developed for stellarator are adapted to RFP. Full 3D equilibrium is reconstructed with VMEC. Analysis of ideal stability (Terpsichore code) started.

Feedback control of MHD stability gives many results. Improvements are obtained by including in control models toroidal geometry and deviations from uniformity of the passive structures. The issue of mode non-rigidity was studied, testing the influence of coil geometry and number. A new integrated simulator for closed loop control experiments was developed and benchmarked. Experimental optimization of tearing modes control is based on a model for the non-linear dynamics of interacting TMs with feedback controlled boundary conditions. RFX-mod was run as a tokamak, with 120 kA plasma current and pulse duration ~ 1.2 s. A current-driven (2,1) RWM is observed as $q_{edge} \sim 2$ in ramped current plasmas. Feedback control stabilizes this mode, and for the first time a feedback stabilized $q_{edge} \sim 2$ tokamak plasma is run without disruptions.

[1] Nat. Phys. 5, 2009.

OV/5-3Rb

Overview of Results in the MST Reversed Field Pinch Experiment

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An overview of MST results is presented. Substantial advances have been achieved in RFP confinement, in the development of auxiliary sources for heating and current drive, and in studies of fluctuations, transport, and magnetic self-organization physics. The density in improved confinement plasmas has been increased above the empirical limit, $n/n_G = 1.5$, using pellet injection. A record density of 0.7×10^{20} m^{-3} has been achieved in high current (0.5 MA) plasmas. Maximum energy confinement is attained at 0.5 MA and lower $n/n_G = 0.13$; this establishes confinement comparable to a same-size, same-current tokamak over MST's full range of plasma current. Experiments using a new 1 MW, 25 keV neutral beam injector are underway, to explore beta limits, energetic particle confinement, momentum transport, and current profile control. Initial NBI results reveal changes in the plasma flow and tearing dynamics. Bulk ion heating is observed. The d-d neutron production evidences good fast ion confinement, but with losses at sawtooth events. Lower hybrid and electron Bernstein wave injection ($\sim$ 100 kW absorbed power for each) are in development for current drive and heating. Electric field and test particle modeling reproduces key features of hard-x-ray emission with lower hybrid, and soft-x-ray emission evidences heating for EBW injection. Results for oscillating field current drive are in good agreement with nonlinear resistive MHD computation, bolstering the OFCD physics basis. Energy confinement with OFCD is measured comparable to that for standard induction. The electrostatic potential is measured in the core of improved confinement plasmas using a heavy ion beam probe. The GYRO code has been modified for RFP equilibria, and ion-temperature-gradient instability is found possible. Non-collisional ion heating generates transient temperatures $T_i = 2 - 3$ keV. The heating efficiency is measured to scale with a fractional power of the ion mass. Broadband magnetic turbulence exhibits a dissipative

nonlinear cascade that could be connected to the ion heating mechanism. A new high rep-rate Thomson scattering diagnostic measures electron temperature fluctuations associated with residual magnetic islands and helical mode structure. The equilibrium and fluctuations of quasi-single-helicity plasmas are investigated with this and MST's other advanced diagnostics.

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OV/5-4

Toroidal Momentum Transport

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Plasma rotation plays a key role in the regulation of turbulence and has a beneficial effect on energy confinement in fusion devices. It can also stabilize resistive wall modes. In reactor plasmas the external torque is relatively small, and for a long time toroidal plasma rotation has been assumed negligible under these conditions. This picture was radically changed by the experimental observation of a large rotation in the absence of an external torque (known as intrinsic rotation). Subsequently, mechanisms of toroidal momentum transport have attracted much attention in the community, leading to a very rapid development in theory and modelling. This paper will give an overview of ALL the developments in this relatively new area and puts them in a global perspective. The emphasis is on theory development and modelling, but experimental validation will also be reviewed. Toroidal momentum transport results from a breaking of symmetry in the direction along the magnetic field. This can be shown at a fundamental level from the symmetry in the gyrokinetic equation, allowing also for the identification of all processes that can generate momentum transport. In the gyrokinetic ordering (expansion in the Larmor radius normalized to the major radius), five processes emerge at lowest order: the finite radial gradient of the rotation, the $\mathbf{E} \times \mathbf{B}$ shearing, the Coriolis force due to the plasma rotation, a finite particle flux in the presence of a toroidal rotation, and the up-down asymmetry in the magnetic equilibrium. At the next order additional effects, such as the parallel velocity non-linearity, appear, only some of which have been identified. The higher order fluxes are small in a reactor plasma, and this paper will concentrate on the leading order effects. The process of momentum transport is complex, and nonlinear simulations play an important role in unravelling the physics, and are necessary for

quantifiable predictions. The paper reports on the extensive analytic / numeric study of the five leading order effects and their interaction. Various experiments that have been undertaken to validate the physics model will be reviewed. This discussion also reveals how the various mechanisms interact to generate the rotation profile. Finally, the paper will discuss the expectations for a reactor based on our present knowledge.

OV — Poster

OV/P-1

The Australian Plasma Fusion Research Facility: Recent Results and Upgrade Plans

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The Australian Plasma Fusion Research Facility will be upgraded to support the development of world-class diagnostic systems for application to international facilities in preparation for ITER. The upgrade will include a materials Diagnostic Test Facility, new diagnostics, heating and vacuum systems, and deliver access to new magnetic configurations relevant to development of edge and divertor plasma diagnostics for next generation devices. New results from some of the optical imaging and magnetic diagnostics underpinning the upgrade plans will be presented, including a innovative technique to synchronously image MHD mode structure using fast optical emission imaging. Initial results provide high resolution radial and poloidal mode structure information in qualitative agreement with a simple cylindrical Alfven eigenmode model. Results of a new multi-view system will be presented, and compared with CAS3D modelling. A new toroidal Mirnov array will supplement the present two 20 channel poloidal arrays, and will provide valuable additional information to our datamining analysis. Our techniques, now applied to several large stellarators and to JT-60U, have been extended to provide automated mode classification of new data, based on maximum likelihood estimates from previously clustered and identified data. Investigation in the influence of magnetic islands on plasma showed several interesting effects, including a local enhancement of confinement under some conditions. Mapping and probe results will be presented.

OV/P-2

Near Term Perspectives for Fusion Research and New Contributions by the Ignitor Program

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The development of fusion reactors continues to call for the achievement of ignition conditions and for the demonstration of plasma regimes suitable for a net power producing reactor. The value of the high field and plasma current density approach of which Ignitor is the most advanced development is confirmed by the recent identification of the attractive characteristics of the so called I-regime on the Alcator C-Mod machine. A near term objective other than ignition that can be pursued in the near term by compact, high density machines is that of producing high flux neutron sources for material testing. Thus, in order to have a usable neutron source, a special machine, whose duty cycle is reasonable and where space is provided to house the materials to be tested, has to be conceived without the goal of ignition. This has been one of the incentives that have led us to consider a new design including the adoption of Magnesium Diboride superconductor with appropriate structural materials in combination with copper for important machine magnet systems (e.g. toroidal). The detailed design of the largest poloidal coils made entirely of MgB_2 is being carried out for the Ignitor machine. Manufacturing of a prototype ICRH antenna is underway, to be installed and tested on the existing full-size sector of the plasma chamber, featuring an innovative plug in system to allow installation and maintenance by the remote handling system. Three experimental campaigns have been carried out at ORNL to test the performance of the Ignitor Pellet Injector, a 4 barrel, two-stage injector designed to reach 4 km/s. At present pellets at 2 km/s have been launched reliably, and new tests are planned for this coming spring.

OV/P-4

Overview of the JT-60SA Project

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The mission of the JT-60SA project is to contribute to the early realization of fusion energy by supporting the exploitation of ITER and research towards DEMO by addressing key physics issues associated with these machines. The JT-60SA will be capable of confining break-even equivalent class high-temperature deuterium plasmas at a plasma current I_p of 5.5 MA and a major radius of ~ 3 m lasting for a duration longer than the timescales characteristic of plasma processes, pursue full non-inductive steady-state operation with high plasma beta close to and exceeding no-wall ideal stability limits, and establish ITER-relevant high density plasma regimes well above the H-mode power threshold. In order to satisfy the plasma performance requirements, the machine has been coherently designed to cover: a wide range of plasma equilibria with divertor configurations covering a higher plasma shaping factor of $S \sim 7$, a lower aspect ratio of ~ 2.5, a high triangularity of ~ 0.5, and a high elongation of ~ 1.9; an inductive plasma current flattop and additional heating up to 41 MW during 100 s; divertor targets to stand up to 15 MW/m^2; negative ion NBI (N-NBI) with high beam energy up to 500 keV; internal coils for RWM stabilization equipped with stabilizing shell; in-vessel components such as divertor cassette to be compatible with remote maintenance. The design integration of the machine was successfully completed in late 2008, so that the project made a large step forward towards its construction, which now foresees the first plasma in 2016. The procurement implementation of the components such as the superconducting magnet system, vacuum vessel, divertor, cryostat and power supply has progressed in both Japan and EU, and manufacturing activities have also commenced with facilities for superconducting coils on the JAEA Naka site and a prototype for vacuum vessel on the manufacturing factory. The development of the JT-60SA research plan has started for future joint exploitation between Europe and Japan in JT-60SA.

OV/P-5

Overview of the ITER Korea Procurement Activities

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The ITER Korea has been established in 2007 as an entity of the Korean Domestic Agency (KODA) that is executing all responsible activities with respect to the ITER Project in Korea. Its main work is designing, manufacturing and delivering on time the ITER in-kind contributions to ITER Organization complying with technical specifications. Procurement activities are in progress on schedule, and some design and R&D activities are being performed to meet the technical specifications for procurement packages. This paper will describe the design and technical activities that currently being conducted in the KODA.

OV/P-6

Progress in Studies of Magnetic Mirror and their Prospects

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The paper describes recent experimental results achieved at GOL-3 and GDT mirror confinement systems. The 12-m-long central solenoid of GOL-3 device consists of 55 single mirror cells with 4.8/3.2 T magnetic field. The plasma density is $2 \times 10^{20} - 5 \times 10^{21}$ m^{-3} at typical electron and ion temperatures up to $1 - 2$ keV. The plasma is heated by injection of high current relativistic electron beam at one end. The new findings in the experiments which suggest significant improvements of the multi-mirror reactor concept are discussed in the paper. Also discussed are the latest results from the GDT experiment, which are obtained after upgrade of the neutral beams.

OV/P-7

Recent Results of T-10 Tokamak

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The poloidal asymmetry, radial correlation lengths and long distance toroidal correlations were investigated by means correlation reflectometry. The concept of the ITB formation at the low order rational surfaces under the conditions of the low density of the rational surfaces and low magnetic shear was proved by the observation of the ITB formation near $q = 1.5$ region with the use of non-central ECRH heating and current ramp up. Heavy Ion Beam Probe diagnostic showed decrease of plasma potential with the rise of density

and its increase under ECRH. Second harmonic EC assisted start up was investigated in T-10. Possibility of the control of runaway electron generation and current decay rate after an energy quench at the density limit disruption was shown using ECRH heating in combination with preprogrammed Ohmic power supply system and gas puffing. Lithium gettering of the limiter and the wall give possibility to reduce significantly impurity level and recycling coefficient up to 0.3.

OV/P-8

The Reconstruction and Research Progress of the TEXT-U Tokamak in China

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The TEXT-U tokamak, having been operated by the University of Texas at Austin in USA, was disassembled and shipped to China in 2004. Now the machine settled down on the campus of Huazhong University of Science and Technology, and was renamed as the Joint TEXT (J-TEXT) tokamak. In spring of 2007, the reconstruction of the J-TEXT tokamak was completed. Subsequently, the first plasma was obtained at the end of 2007. Accompanying development of the routine diagnostic systems, experimental study focused on MHD activity, density limits, disruption events, and fluctuations and turbulence in the plasma edge region. At present, the limiter configuration of the J-TEXT tokamak is fixed at a major radius $R = 1.05$ m and a minor radius $a = 0.27$ m. And the typical Ohmic discharge achieved in the machine is the production of plasma current of more than 200 kA with a flattop duration of 300 ms at a center-line toroidal magnetic field $B_T = 2.2$ T. Normally, the line averaged density of such discharges exceeds $\sim 2 \times 10^{19}/\mathrm{m}^3$, while the electron temperature is near 700 eV.

The reconstruction of the J-TEXT tokamak not only included the assembly of the vacuum chamber, magnetic coils, and iron core transformer, but also involved redesign and manufacture of pumping stations and gas puffing system, restructure of magnetic field power supplies, setup of a new water cooling and de-ioning system, data acquisition and management system, and so on. Moreover, a new discharge control system, which was developed on the base of QNX commercial real-time operation systems, was implemented for J-TEXT operation.

Meanwhile, some diagnostic systems used to facilitate the routine operation and research scenarios were developed on the J-TEXT tokamak, such as 2D magnetic probes, Langmuir probes, 7-channel HCN interferometer, 2 mm microwave interferometer, soft X-ray arrays and AXUV arrays for cross-section tomography, soft X-ray pulse-height analysis, and hard X-ray detectors, etc. Benefiting from these diagnostic tools, the observation of MHD activity and the statistical analysis of the disruption events were done. Measurements of the electrostatic fluctuations in the edge region and conditional analysis of the intermittent burst events near the LCFS were also made. The preliminary results from the J-TEXT tokamak were described in detail.

EX
Magnetic Confinement Experiments

EXC — Oral

(*Magnetic Confinement Experiments: Confinement*)

EXC/1-1

High Confinement Hybrid Scenario in JET and its Significance for ITER

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In the recent 2008-2009 campaigns, JET has achieved a comprehensive exploration of the plasma scenario, known as the "hybrid" scenario. Confinement improvement up to $H_{98(y,2)} = 1.4$ have been achieved by broadening the current density profile using current ramp down techniques prior to the main heating phase. This technique has been applied successfully to high triangularity ITER-like shape and thus to higher density (75% of the density limit) with $H_{98(y,2)} = 1.3 - 1.4$, $\beta_N \sim 3$ and has been maintained for more than 6s (about one τ_R).

Simulation with the CRONOS code can reproduce the effect of the current ramp down and confirm that the pulse the q profile shows a large region of flat shear with q_{min} close to 1.

The occurrence of 4/3 and then 3/2 modes are also often observed during the current diffusion phase. Stability analysis of the 2/1 and 3/2 NTMs has shown that the critical island width lowers as the plasma current diffuses towards the centre implying that broad current density profile is a necessary condition to keep the discharge in stable conditions.

Plasma wall clearance appears to minimise recycling processes and shape changes as the internal inductance decreases can also affect the pedestal confinement, implying that shape is an important element in the optimisation of confinement. The pedestal contribution to the global confinement has also been specifically investigated in dedicated input powers and q_{95} scan of discharges.

In the plasma core, direct influence of the q profile on the underlying ITG turbulence with toroidal rotation is invoked as a possible mechanism for the reduced transport in the plasma core.

The operational space of these improved confinement hybrid plasmas has been extended to different normalised parameters such as q_{95} and normalised gyro-radius ρ^* from 3×10^{-3} to 2.1×10^{-3} at fixed shape and β_N. Finally, the lowest ρ^* achieved in JET has been extrapolated to ITER by making a dimensionless scaling of kinetic profiles in ρ^*, shape and rotation. From the 0D analysis, this scenario could reach an estimated fusion gain factor Q of 7 which uncertainties can be now ascertained more accurately as function of ρ^* and shape.

EXC/1-3

I-Mode: An H-Mode Energy Confinement Regime with L-Mode Particle Confinement on Alcator C-Mod

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It is highly desirable to understand, and separately control, the energy and particle transport channels in magnetically confined plasmas in order to meet the multiple, and sometimes conflicting, requirements of energy confinement, ash removal, impurity control and pedestal stability. An improved energy confinement regime, I-mode, is studied on Alcator C-Mod, a high-field, high-power density divertor tokamak with an all-metal wall. I-mode features an edge energy transport barrier without an accompanying particle barrier, leading to several performance benefits: H-mode $H_{98}\sim1$ energy confinement, avoidance of Edge Localized Modes, a reduction of impurity radiation with a high-Z metallic wall, and density control using divertor cryopumping. I-mode is a confinement regime that appears distinct from both L-mode and H-mode, combining the most favorable elements of both, but being neither. The I-mode regime is obtained on Alcator C-Mod with ion grad-B drift away from the active X-point and using ion cyclotron RF heating. The transition from L-mode to I-mode is primarily identified by the formation of a high temperature edge pedestal, whilst the edge density profile remains nearly identical to L-mode. The signatures of L to I-mode transitions, such as increasing edge temperature, stored energy and fusion reactivity, vary substantially in their abruptness, unlike the sudden transitions found for H-mode onsets. As in L-mode, SOL and divertor density profiles remain broad which should be beneficial for power handling, impurity shielding and ICRF coupling. Laser blowoff injection shows that I-mode core impurity confinement times are nearly identical with L-mode, despite the enhanced energy confinement. These features of weak particle confinement appear to be responsible for lowering high-Z metal accumulation in the core plasma, which often degrades energy confinement in ICRF-heated H-modes with uncoated high-Z metal walls. I-mode allows access to stationary collisionless pedestals ($\nu^*\sim0.1$) with ~1 keV pedestal temperature. This regime also features a high-frequency ($f\sim100-300$ kHz) edge/pedestal density fluctuation which is weakly coherent ($\Delta f\sim50$ kHz). This fluctuation may participate in energy and particle transport regulation in the edge.

EXC/1-4

Towards a Steady-State Scenario with ITER Dimensionless Parameters in JET

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Plasmas suitable for ITER Steady-State scenario must have $\beta_N = 3$ ($> 80\%$ from thermal particles), $H_{98} = 1.4 - 1.5$ and bootstrap current fraction (f_{BS}) = 50%. In addition, it is important to obtain dimensionless parameters such as the normalised Larmor radius (ρ^*), collisionality (ν^*), ratio of electron to ion temperature (T_e/T_i), q-profile and Mach number as close as possible to those in ITER since they affect transport and/or current drive. This motivated experiments in JET where the performance of high triangularity NBI-only plasmas developed to study stability and confinement at high β_N and H_{98} was extended to higher B_T, I_P (2.7 T, 1.8 MA, $q_{95} = 4.7$) and power (electron heating ICRH, LHCD in addition to NBI). Good ICRF coupling is obtained with gas dosing, with good edge confinement ($H_{98} \sim 1$) and type I ELMs. With the ICRF ELM resilient systems, up to 8 MW is coupled. LH coupling is affected by the nearest ICRH antenna and PLH< 3 MW. The following steady ($> 10 \times \tau^E$) and peak performances are obtained: $H_{98} = 1.2$ (up to 1.35), $\beta_N = 2.7$ (up to 3.1). The thermal fraction is up to 80% and $\langle T_e\rangle/\langle T_i\rangle = 0.9 - 0.95$, with $T_e = T_i$ at the pedestal. The Mach number at mid-radius =0.4, and $\rho^*/\rho^*_{ITER} = 2$, $\nu^*/\nu^*_{ITER} = 4$, all lower than in previous, lower B_T, experiments. In the range of target q-profiles used ($1.8 < q_{min} < 2.9$), only plasmas with an internal transport barrier, in addition to an edge transport barrier, have $H_{98} > 1.1$. In some shots, $n = 1$ kink modes are seen, often limiting the performance and in a few cases causing disruptions. The role played by the pressure gradient and fast particles in destabilising the mode is under investigation. In all shots the q-profile is evolving, indicating a shortage of non-inductive (NI) current. Interpretative modelling show that $f_{BS} = 40-46\%$ where the highest f_{BS} is for plasmas with highest q_0 and β_N. Predictive modelling shows that f_{BS} goes from 46% to 50% when scaling to ITER ν^*. NBI provides $\sim 20\%$ of externally driven NI current, but LHCD contributes little, because the LH waves can not penetrate past the pedestal for the B_T and density of these experiments. An ECRH system was recently proposed for JET. Predictive modelling shows that ECCD can provide the narrow off-axis current required to maintain the q-profile favourable to the high performance.

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EXC/1-5

Progress of Superdense Plasma Research in LHD: Sustainment and Transport Study

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The high density operation is favorable for the fusion reactors since the fusion reaction rate is proportional to the density squared. In addition, the energy confinement is also expected to improve with density in helical and tokamak devices. In tokamaks, however, the density is limited by the well-known Greenwald density. On the other hand, in net current free helical systems, there is no essential density limit like tokamaks. If anything, it is only limited by the power balance with edge radiation. Thus high density operation research is intensively conducted in helical systems. In 2005, the superdense core (SDC) mode with central electron density of more than $\sim 1 \times 10^{21}$ m^{-3} was discovered in LHD with the formation of an internal diffusion barrier (IDB) when a series of pellets was injected into the neutral beam heated plasma under the condition of low edge neutral pressure. Recently the IDB-SDC plasma with central electron density of $\sim 2 \times 10^{20}$ m^{-3} could successfully be sustained for more than 3 seconds. The IDB was robustly maintained against the strong perturbation of externally injected fuelling pellets. No serious increase in radiation power or impurity accumulation was seen during the discharge. In such a high density regime, the large Shafranov shift takes place which strongly modifies the edge magnetic topology. Therefore it is interesting to investigate the transport properties in the modified edge region which surrounds the SDC plasma. Temporal behavior of electron density profiles was analyzed to obtain the diffusion coefficient for particles. It was clearly shown that the diffusion coefficient in the edge region is larger than that in the core region. The resonant magnetic perturbation (RMP) was applied to further modify the edge magnetic topology. In the discharge with RMP, reduction of the edge density was observed, which is called "particle pump-out" also observed in tokamaks. Transport analyses surely indicate the increase in the diffusion coefficient in the edge region. These results suggest that modification of magnetic topology in the edge region enhances the particle transport.

EXC/2-1

Scaling of H-Mode Pedestal and ELM Characteristics in the JET and DIII-D Tokamaks

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Because of the influence of the H-mode pedestal height on overall performance, and of ELM size on the survivability of plasma facing components, an understanding of the scaling of these features with machine size and other quantities is desirable for design and feasibility studies of future tokamaks such as ITER. The dependence of the pedestal height and ELM size on ion gyroradius, $\rho^* = \rho/a$, is important since ρ^* in ITER will be smaller than existing tokamaks, and theories based on the $\mathbf{E}\times\mathbf{B}$ velocity shear suppression of ion scale turbulence suggest the width of the edge transport barrier should decrease as $(\rho^*)^{1/2}$ or even $(\rho^*)^1$. The effect of ρ^* was studied in experiments exploiting the size difference of JET and DIII-D to achieve a factor of 4 change in pedestal ρ^* keeping other dimensionless parameters fixed. Little dependence was found in the width of the edge steep gradient region of the T_e profile on ρ^*, where a scaling of $w_{T_e} \propto (\rho^*)^x$ with $x \geq 0.25$ was ruled out to a 95% confidence level, and to 80% confidence for $x \geq 0.1$. ELM size decreased strongly with decreasing ρ^* in DIII-D but this trend saturated at the smaller ρ^* in JET. The plasma density played a role in the structure of the edge density profile consistent with the effects of neutral penetration depth on the edge particle source. Higher density discharges had density pedestals shifted outward relative to the temperature profiles resulting in a narrowing of the pressure width. A combined data set of JET and DIII-D discharges had ETB widths and pedestal heights in agreement with the predictions of the EPED1 model. EPED1 is based on a combination of kinetic ballooning and peeling ballooning mode stability criteria and predicts a width scaling with pedestal poloidal β as $(\beta_{POPL})^{1/2}$. A broader database for β scaling will be presented. The effect of collisionality on pedestal structure was consistent with its effects on peeling-ballooning stability in the EPED1 model giving a reduction in ETB width and pedestal height when the collisionality is high enough to suppress the pedestal bootstrap current.

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EXC/2-3Ra

L/H Transition and Pedestal Studies on MAST

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The pedestal performance is pivotal for the success of ITER. Here, we present recent investigations on MAST focussed on L/H transition physics, pedestal transport and pedestal stability. The profile evolution of edge parameters such as the electron temperature (T_e) and density (n_e), the ion temperature (T_i, using a novel technique), the radial electric field (E_r) and the toroidal edge current density (j_{tor}, using the Motional Stark Effect) has been studied with a time resolution as fast as 0.2 ms in the case of T_e, n_e and E_r. The species dependence of the L/H power threshold, $P(LH)$, was measured by comparing D and He plasmas. On MAST, $P(LH)$ in He is about 50% higher than in D. At same heating powers above $P(LH)$ H-modes in He have a slightly narrower pedestal than in D. If the distance between the X-point and the target plates is reduced $P(LH)$ is lowered even in double null. The data in L-mode prior to H-mode suggest that neither the gradient nor the value of the mean T_e or E_r at the plasma edge play a major role in triggering the L/H transition. This is supported by observations of E_r and T_e made during suppression of the L/H transition with resonant magnetic field perturbations. Following the L/H transition the fluctuations are suppressed on a 5 to 10 times faster time scale than the pedestal is formed. Owing to the conductive and convective heat transport, compared to the convective particle transport, ne evolves faster than T_e. In both L- and H-mode the edge T_i profiles are flat with $T_i > T_e$ in the pedestal region reaching ~ 150 eV in H-mode close to the separatrix. A clear increase of jtor by a factor of ~ 5, of similar magnitude to the calculated bootstrap current, is observed when changing from L- to H-mode. These measurements on a ~ 5 ms time scale allow us for the first time in a ST to calculate the peeling-ballooning stability in the edge with all profiles measured. This edge stability picture used to explain edge localised modes (ELMs) breaks down for high confinement H-modes obtained with counter current NBI on MAST, where large ELMs expelling up to 7% of the plasma energy are observed despite the shallow pedestal gradients. These H-modes show an edge mode similar to the edge harmonic oscillation that suppresses the ELMs in quiescent H-mode, but without the strong well in E_r and ELMs being still present.

Work supported by UK EPSRC and EURATOM.

EXC/2-3Rb

L-H Threshold Studies in NSTX

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Recent experiments in the low aspect ratio National Spherical Torus Experiment (NSTX) have been run in support of the high priority ITER and ITPA issue of access to the H-mode. Specifically, a series of L-H threshold experiments have addressed the effect of plasma ion species, applied 3D fields, wall conditioning, plasma current and plasma shape/X-point position on the L-H power threshold and local parameters leading up to the transition. Experiments on the species effect revealed that the L-H threshold power for deuterium and helium were comparable, but that the conclusion depended on how the threshold power was defined. There was a $\sim$ 35% reduction in the threshold power normalized by line-averaged density for discharges using lithium evaporation to coat the plasma facing components than for those that did not. Application of largely non- resonant $n = 3$ fields at the plasma edge, potentially critical for suppression of ELMs in ITER, resulted in about a 65% increase in density-normalized threshold power even with no change in plasma rotation. The plasma current also has a controlling factor in the L-H transition in NSTX, with normalized threshold powers almost a factor of two greater at 1 MA than at 0.7 kA, consistent with calculations from XGC0 showing a deeper E_r well and stronger E_r shear near the edge at lower current. Experiments based on XGC0 predictions indicated that low triangularity discharges required the lowest auxiliary heating power to transition into the H-mode, although these results also depend on the precise definition of threshold power. To within the constraints of temporal and spatial resolutions, no systematic difference in T_e, n_e, P_e, T_i, v_ϕ or their derivatives was found in discharges that transitioned into the H-mode versus those at slightly lower power that did not. Finally, it was found that RF-heated discharges could attain values of $H_{98y,2} \sim 1$ in ELM-free conditions for powers just above the power threshold. NBI heated, ELM-free H-modes could also achieve $H_{98y,2} \sim 1$, but only after $\sim$ 50 ms after the L-H transition.

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EXC/2-4Ra

L-H Transition Studies on DIII-D to Determine H-Mode Access for Non-Nuclear Operational Scenarios in ITER

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A comprehensive set of L-H transition experiments has recently been performed on DIII-D to determine the requirements for access to H-mode plasmas in ITER's first (non-nuclear) operational phase with H and He plasmas and the second (activated) operational phase with D plasmas. The results from these experiments have revealed that the H-mode power threshold, P_{TH}: (a) increases with the applied torque; (b) increases in the presence of $n = 3$ resonant magnetic perturbations; (c) can be significantly reduced by changing the plasma geometry; (d) shows a dependence on the edge toroidal rotation; (e) is not significantly affected by the application of magnetic error fields, expected from a mock-up of the test blanket module assembly to be used in ITER, for either ECH or NBI heating in D plasmas.

New results with near pure He plasmas ($> 90\%$ purity) indicate that the H-mode power threshold in He plasmas is about $30\% - 50\%$ higher than D plasmas depending on the applied torque. No systematic variations in the edge T_e and T_i are observed with the H-mode power threshold. However, a significant variation of P_{TH} is seen with the edge toroidal rotation, v_ϕ. The main reason for this effect appears to be the influence of the toroidal rotation on the edge radial electric field, E_r, from force balance. The presence of a large v_ϕ positive component competes against the negative diamagnetic term and hence reduces the shear in the edge E_r, and makes it harder to attain the necessary conditions for turbulence suppression at large toroidal rotation.

The H-mode power threshold increases with the application of the I-coil current, which is normally used to induce $n = 3$ resonant magnetic perturbations required for edge localized mode (ELM) suppression. The I-coil current is applied during the L-mode prior to the main heating phase and P_{TH} is observed to increase significantly even at low I-coil currents of 2 kA, whereas the normal range of I-coil current for ELM suppression is 4 kA and above. This behavior indicates that, for ITER applicability, operational procedures have to be developed and tested in which the I-coil current is zero or very low during the H-mode transition, and is ramped up after the transition to the nominal values required for ELM suppression before the first ELM.

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EXC/2-4Rb

JET ^{4}He ELMy H-mode Studies

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This paper reports H-mode plasma results from the 2009 JET ^{4}He campaign which, as part of a wider ITPA study, have extended the physics basis to enable improved predictions of ^{4}He, H-modes in ITER. L-H threshold experiments included the first ever dedicated study of the effect of concentration on the L-H threshold. ^{4}He concentration was varied from 1 to 87% and was found to have little impact on the power threshold. This is in line with recent ASDEX Upgrade studies, but in contrast to JET 2001 results, which found that ^{4}He plasmas have a 40% higher threshold than D equivalents. A study of the density dependence of the L-H threshold power in ^{4}He and D found very different behaviour which may, in part, explain the differences between the JET 2001 and 2009 studies which were performed at different densities. The 2009 studies included the first experiment to measure the threshold required for Type I ELMs in ^{4}He plasmas. A series of ^{4}He and D plasmas with matched field (1.8 T), current (1.7 MA), shape (triangularity of 0.4) and divertor configuration were performed with different input powers. Similar Type I ELM thresholds were found for the D ($6.7-9.3$ MW) and ^{4}He ($7.5-9.3$ MW) plasmas. By normalising to a standard L-H threshold scaling, these results can be extrapolated to the ITER, ^{4}He, half-field, baseline conditions (2.65 T, 7.5 MA and density of 50% of the Greenwald limit) where they predict a required input power of $42-48$ MW, or $23-86$ MW for an appropriately chosen 95% confidence interval. These intervals are largely consistent with the design auxiliary heating capacity of ITER. Confinement and ETB studies found that energy confinement times in ^{4}He plasmas were approximately 60% of those for reference D ones, in line with previous studies on several machines. The relative impact of the core transport and ETB on this difference will be assessed and reported on. IR camera measurements during He4, Type I ELMs show that the heat load is deposited over significantly longer periods than for D equivalents. The impact of magnetic perturbations on Type I ELMs in He^4 was also studied. All of these results will

be compared with those from other machines and the combined physics basis will be used to make predictions for ITER.

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EXC/2-5Ra

H-mode in Helical Devices

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The classical H-mode occurs in 3D magnetic configurations, Stellarators (Wendelstein 7-AS), Heliotrons (LHD, CHS, Heliotron-J) and Heliac configurations (TJ-II) with a phenomenology close to the observations in 2D. The ELM phenomenology depends sensibly on the magnetic configuration which opens a 3D-specific opportunity for ELM mitigation with possible merit for a reactor concept.

In helical devices where plasma $\mathbf{E}\times\mathbf{B}$ rotation is already predefined by the ambipolarity condition of the convective radial fluxes an edge transport barrier with strong mean $\mathbf{E}\times\mathbf{B}$ flow can develop on a profile timescale without a sudden suppression of turbulence. In contrast the classical H-mode shows a fast bifurcation character demonstrated by the existence of forward- LH and HL back-transitions, a jump in turbulence amplitude and a sudden change in the oscillations of the turbulence propagation velocity and its radial gradients. The latter may display zonal flow structures and supports the paradigm that at the transition turbulent transport-carrying vortices are suppressed by fluctuating sheared flow structures on top of the developing mean shear flow.

Threshold parameters depend on the 3D magnetic configuration: In devices characterized by a strong magnetic shear with sign opposite to a tokamak the transition is achieved for particular edge configurations e.g. characterized by an edge island or an ergodic layer by exceeding a threshold power. In contrast low shear devices reach the LH transition at constant heating power by increasing the density under conditions at or even significantly below the tokamak H-mode power threshold. Their transition depends critically on the edge rotational transform which governs the presence of edge islands and rationals right in the relevant gradient region. The influence of these elements on the spin-up of flows is discussed. As a working hypothesis it is supposed that the fast bifurcation to a high-rotation state may be biased (or damped e.g. in the case of an ergodic layer) by the specific conditions for flows in the 3D magnetic configuration and by the mean $\mathbf{E}\times\mathbf{B}$ shear flow resulting already from the ion-root conditions.

EXC/2-5Rb

Overview of the Construction and Scientific Objectives of the Wendelstein 7-X Stellarator

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The main objective of the Wendelstein 7-X stellarator, presently under construction in Greifswald, is to demonstrate the integrated reactor potential of the optimized stellarator line. An important element of this is the achievement of high heating-power, high confinement in fully controlled steady-state operation. Such a reactor relevant steady-state operation has not yet been demonstrated in a stellarator and represents the major scientific goal of Wendelstein.

Construction of Wendelstein 7-X has progressed considerably over the last years. All superconducting coils and all large cryostat components (plasma vessel, ports, outer vessel) have been delivered. Fabrication of the bus-bars, of the cryo-pipes and of the in-vessel components is according to plan.

The main objective of the project has shifted from component design and manufacturing to device assembly. The assembly strategy foresees the assembly of the five identical modules in several steps. As of now, all five modules are being assembled simultaneously (at different completion stages).

After completion of the construction and commissioning of the device, a first operation phase with short pulse duration ($5-10$ s at 8 MW heating power) is scheduled to investigate stellarator optimisation and to develop an integrated steady-state operation scenario that is compatible with the island divertor concept. After this operation, the inertially cooled test divertor will be replaced by an actively cooled divertor, designed for stationary heat fluxes of up to 10 MW/m^2.

The main tool for heating and current drive will be an ECRH system with 10 gyrotrons, delivering up to 10 MW over 30 minutes. The other heating method for this period is a neutral beam injection system with 2 beam-lines, providing a heating power of 7 MW hydrogen for up to 10 s. The diagnostic system needs as well dedicated developments for steady state operation.

The paper will describe the main components of the device and an overview of the construction progress. The physics approach to achieve high performance steady state operation will be described, based on the foreseen heating and diagnostic tools. Particular emphasis is put on the research programme of the initial operation and its scientific objectives ranging from an assessment of the optimization criteria to the preparation of high power steady state operation.

EXC/3-1

JET Rotation Experiments towards the Capability to Predict the Toroidal Rotation Profile

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Plasma rotation and momentum transport are currently very active areas of research, both experimentally and theoretically. An inward momentum pinch, in many cases very significant in size, has been reported on JET. The pinch number has been found to depend most strongly on density gradient length, a factor of 2 increase in the density gradient length yielding roughly a factor of 1.5 increase in the pinch number. It is also increased by about $20 - 30\%$ when the collisionality increases a factor of 3. Intrinsic rotation has been investigated with ripple levels ranging from 0.08% to 1.5%. Both in the case of plasmas with no momentum input (ICRH) and without any fast ions (Ohmic), increasing ripple was found to cause counter rotation. Intrinsic rotation experiments with large ICRH power (H-mode) and at normalised β, $\beta_N = 1.3$ showed that the plasma rotation was very close to zero on JET. An experiment to study Mode Conversion Flow Drive (MCFD) was performed where strong counter-rotation up to 30 km/s was observed. Concerning rotation sink studies, a strong toroidal rotation braking has been observed in plasmas with pplication of an $n = 1$ magnetic perturbation field on JET. The braking torque was found to be in a fairly good agreement with the predicted Neoclassical Toroidal Viscosity torque. Another type of momentum sink studies concerned plasma toroidal angular momentum and thermal energy losses by Edge Localised Modes (ELMs). The analysis shows a consistently larger drop in momentum in comparison to the energy loss associated to the ELMs. This difference, in combination with the observed longer build up time for the momentum density, results in a significant drop in thermal Mach number at

the edge. Combining the new results from the core and edge momentum transport (pinch dependencies, relation to ion heat transport and ELM losses), intrinsic rotation sources (H-mode at higher beta and MCFD) and magnetic perturbation (ripple and $n = 1$) is a significant step forward in understanding the physics of toroidal rotation. This also creates a more robust and more reliable base to predict the toroidal rotation profile for example in ITER.

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EXC/3-2

Core and Edge Toroidal Rotation Study in JT-60U

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Relation between the toroidal rotation velocities (V_t) in the core and edge regions has been investigated in H-mode and internal transport barrier (ITB) plasmas with small external torque input from the viewpoint of momentum transport. Main results are as follows: (i) The core-V_t varies with a transport time scale after a rapid change in the edge-V_t at L-H transition. In steady-state, a linear correlation between the core- and edge-V_t is observed in H-mode plasmas with the ion pressure gradient (∇P_i) being small, which can be explained by the momentum transport. (ii) The V_t profiles with the ∇P_i being large have been reproduced by considering a residual stress term, which is the product of the gradPi multiplied by the momentum diffusivity, proposed in this paper. (iii) The V_t inside the ITB is insensitive to the perturbation applied in the edge region and the data, which show the confinement improvement of momentum inside the ITB, has been obtained.

EXC/3-3

Progress towards a Physics based Phenomenology of Intrinsic Rotation in H-Mode and I-Mode

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This paper presents progress toward an improved, physics based phenomenology of intrinsic rotation expressed in terms of local parameters. Specifically, we show that central intrinsic rotation speeds in C-Mod H-mode plasmas scale with the edge pressure gradient. We directly demonstrate the existence of a fluctuation-induced residual stress which drives intrinsic rotation in a basic plasma experiment. Finally, we formulate and calculate a measure of the efficiency of intrinsic rotation in heat flux driven systems.

EXC/3-4

On Plasma Rotation with Toroidal Magnetic Field Ripple and no External Momentum Input

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In a fusion reactor or ITER, the plasma heating will be mainly provided by alpha particles, which does not impart a net torque on the plasma. Another issue is related to the Toroidal magnetic Field ripple ($< 0.5\%$ with inserts), whether or not the ripple induced particle fluxes could have a non negligible effect on plasma rotation and how this effect competes with turbulence driven rotation. Consequently, understanding plasma rotation with low momentum and its relationship to TF ripple and associated neoclassical effect is particularly important.

Tore Supra is a large size tokamak with no external momentum input and a strong magnetic field ripple (up to 7% at the plasma boundary), which makes the device well suited for addressing ripple issues. Radial electric field measurements have been found to agree

well with ripple-induced neoclassical predictions in the outer part of the plasma, suggesting that the main contribution in the ambipolarity condition comes from ripple induced particle fluxes. This raises the issue of the toroidal and poloidal velocity behaviours, and to which extend they are described by neoclassical predictions. We show that in ohmic L-mode plasmas with ripple amplitude at the plasma boundary varied from 0.8% to 7%, the toroidal rotation increases in the counter-current direction with the ripple strength, suggesting a thermal particle ripple loss mechanism. The radial electric field is not found to vary much with the ripple strength while the poloidal rotation, inferred from the radial force balance equation, is found to increase slightly at the plasma edge. Toroidal and poloidal velocities are compared with thermal loss neoclassical predictions including ripple trapping effects and plasma collisionality. RF L-mode heated plasmas (high ripple case) are also investigated. Here, fast particle effects add to the neoclassical ripple friction, and the toroidal velocity profile peaking is found to increase with the injected ICRH power in the counter-current direction. Adding LH power leads to a dramatic change in the toroidal rotation velocity profile with a velocity increment in the co-current direction, which is consistent with JET but opposite to C-Mod observations. Those ICRH and LHCD plasma observations qualitatively agree with a fast ion (resp. electron) ripple loss mechanism, although another mechanism can not be excluded.

EXC/3-5

Characterization of the Effective Torque Profile associated with Driving Intrinsic Rotation on DIII-D

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Recent experiments on DIII-D have focused on elucidating the drive mechanisms for intrinsic rotation in tokamak fusion plasmas. At the edge ($\rho > 0.8$) of a wide range of DIII-D H-mode plasmas, a clear scaling of the effective torque associated with the intrinsic rotation is observed with the edge pressure gradient. For plasmas such as these, with nominally zero toroidal rotation, the radial electric field arises predominantly from the pressure gradient. Therefore, the observed dependence of the edge "intrinsic torque" on the pressure gradient is consistent with such models relating the $\mathbf{E} \times \mathbf{B}$ shear as an important drive for intrinsic rotation. If the edge pressure gradient is responsible for the residual stress leading to a spin up of the plasma from rest, then the H-mode pedestal may provide a ubiquitous mechanism for providing intrinsic rotation in fusion plasmas, and may help to explain the commonality of the observed H-mode intrinsic rotation across multiple machines. The intrinsic torque in the core ($\rho < 0.5$) of H-mode plasmas appears to be much more complex, although also generally much smaller than observed in the edge. Even though the core intrinsic torque tends to be small, some cases have been found where it is large enough to modify the rotation profile. For example, in plasmas with electron cyclotron heating (ECH) deposited near axis, a significant counter intrinsic torque has been observed in the inner region of the plasma, unlike typical H-modes where the core intrinsic torque is essentially zero. Large intrinsic torques have also been observed in the core of quiescent H-mode (QH-mode) plasmas. In these QH-mode plasmas, the net result when integrated across the whole profile is an intrinsic torque that is in the counter I_p direction. Recent studies of the residual stress with the global gyrokinetic code GYRO, have uncovered a novel result; namely that nonlocal profile variations appears capable of generating large residual stresses and associated momentum flows.

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EXC/6-3

FTU Results with the Liquid Lithium Limiter

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On FTU, experiments are in progress with a liquid lithium limiter (LLL) with a capillary porous system (CPS) to investigate important physical and technological issues related to the use of liquid metals as plasma facing materials.Lithization has led to a strong reduction of heavy impurities and of oxygen concentration inside the plasma as well as of particle recycling from the walls due to the pumping effect of lithium. These results have produced strong modifications on plasma characteristics such as: Z_{eff} in ohmic discharges from 1.5 at low density ($= 0.5 \times 10^{20}$ m^3) to 1.0 at higher density ($> 1.0 \times 10^{20}$ m^3), radiation losses lower than 30% of the input power and electron temperature in the SOL a factor 1.5 higher than for boronized and fully metallic discharges. The analysis of an extensive database of boronized and lithized plasma show that extremely peaked profiles are obtained in presence of MARFE in both cases, but at very high density only in lithized discharges. The profile evolution of density clearly shows a strong steepening at $r/a = 0.8$. The central density increases strongly, while the peripheral density for $r/a > 0.8$ remains nearly constant. Similarly, the small increase in the SOL density from Langmuir probes is well below that expected from the scaling law Holding for FTU, according to which $n_{e,SOL} \propto \bar{n}_e^{1.46}$. An analysis of these plasmas has been carried out by means of a gyrokinetic code to establish the role of collisionality and of the density gradients on the observed phenomenology. The confinement time dependence on density of lithium discharges follows the general trend of the old FTU data base with a transition from linear (LOC) to saturated (SOC) regime but displaying a higher saturation level of 70 ms instead of the former 50 ms. First experiments with D_2 pellets injection in ohmic lithized plasmas ($I_p = 0.5$ MA, $B_T = 6$ T) show an increase by a factor 2 of the electron density relaxation time after pellet injection in comparison with not lithized discharges. From the technological point of view, very good indications were obtained from the use

of LLL under high heat fluxes, up to 5 MW/m^2, by shifting the plasma directly on the LLL surface for about 300 ms. In these experiments, a further progress in technology has been made by changing stainless steel with tungsten as material for the matt wires of CPS structure.

EXC/7-1

Behaviour of Mean and Oscillating $\mathbf{E} \times \mathbf{B}$ Plasma Flows and Turbulence Interactions during Confinement Mode Transitions

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Turbulence driven oscillating $\mathbf{E} \times \mathbf{B}$ plasma flows - geodesic acoustic modes (GAMs) - affect edge transport in magnetic confinement devices via enhanced velocity shearing of turbulent eddies. The GAM is universally observed in low confinement L-mode regimes, but not in the high confinement H-mode. The interaction of the plasma turbulence, mean and oscillatory $\mathbf{E} \times \mathbf{B}$ flows is investigated across the L to H-mode transition in the ASDEX Upgrade tokamak using Doppler reflectometry and slow heating power ramps. At low collisionality $\nu^*_{ped} \sim 1$ conditions the L-mode first transitions into an intermediate improved confinement state with an enhanced average edge E_r well and energy confinement, but without ELM activity. Both the edge flow and density turbulence also rise and pulsate $(2-4$ kHz) with a limit cycle behaviour between high and low E_r phases. A strong GAM is still present during the high E_r phase, but with a reduced radial extent coincident with a narrower E_r well. With increasing input power the GAM displacement amplitude grows, before a second transition into a well-developed H-mode occurs with the formation of a clear temperature profile pedestal and reduced edge density turbulence. The turbulence pulsing decays and the GAM is replaced by large amplitude broadband random flow fluctuations. It is speculated that the GAM enhancement may provide the trigger for the L-H transition.

EXC/7-2

Multi-scale/Multi-field Turbulence Measurements to Rigorously Test Gyrokinetic Simulation Predictions on the DIII-D Tokamak

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This presentation describes the overall progress in rigorously testing gyrokinetic turbulence simulation through a series of carefully designed experiments performed within the Transport Mode Validation Task Force at DIII-D. These experiments take full advantage of the diagnostic, heating, and plasma control capabilities available at DIII-D which provide significant constraints for testing and validating gyrokinetic simulations. A unique array of multi-field, multi-scale turbulence measurements has been utilized to study a range of target plasmas with excellent spatial coverage ($r/a \sim 0.55 - 0.85$). New measurements include local, wavenumber-resolved TEM-scale $\tilde{n}$, turbulence flows, density-temperature fluctuation crossphase, as well as previously available ITG and ETG scale n_e and low-k T_e fluctuations. New results include: 1) Agreement between n_eT_e crossphase measurements and gyrokinetic predictions when ECH heating is used to vary T_e; 2) Tests of electron mode physics by making ITG modes sub-dominant to electron modes (TEM and ETG) show that intermediate TEM/ETG-scale $\tilde{n}$ increases with increasing a/L_{Te} as predicted. However, higher a/L_{Te} levels show a surprising and unpredicted opposite response; and 3) In elongation (K) scans the measured thermal transport and low-k fluctuations (n_e and T_e) decrease with increasing k consistent with general predictions. These comparisons, although still in the early stages, are indicating some general observations. Predictions of transport and fluctuation levels in the middle core region ($0.4 < r/a < 0.75$) are often in better agreement with experiment than those in either the outer region ($r/a \geq 0.75$), where edge effects may become important, or with those in the inner region ($r/a \leq 0.4$) where some of the scale length assumptions in the gyrokinetic prescription may become less reliable. Further, simulations of plasmas which are not dominated by low k turbulence require extension to successively higher k ranges perhaps indicating the need for a fully coupled low through high k simulation. In summary, careful comparison of experiment and simulation has improved the design of rigorous validation experiments as well as identifying the critical importance of utilizing accurate synthetic diagnostics.

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EXC/7-3

Experimental Study of Zonal Flow, Geodesic Acoustic Mode and Turbulence Regulation in Edge Plasmas of the HL-2A Tokamak

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The spectral characteristics of low frequency zonal flows (LFZFs) and geodesic acoustic modes (GAMs) were studied using Langmiur probe array set in the HL-2A tokamak edge plasmas with Ohmic and electron cyclotron resonance heating (OH and ECRH). The intensity of the LFZFs is observed to increase and decrease with increases of ECRH power and safety factor q, respectively, while the intensity of the GAMs increases with ECRH power as well as q. The radial profiles and propagation of the LFZFs and GAMs were also analyzed and compared for Ohmic and ECRH plasmas. Furthermore, the spatio-temporal (S-T) structures embodied in the ambient turbulence (AT) and resulting from its interaction with zonal flows are found similar to those in LFZFs and GAMs. The results of the present experiment provide the first concrete evidence that zonal flow indeed regulate the AT in tokamak plasmas.

EXC/7-4Ra

Radial Structure of Fluctuation in Electron ITB Plasmas of LHD

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Meso- and macro-scale fluctuation structures are discovered using a correlation technique of ECE in the LHD plasma with an internal transport barrier. Characteristics of the fluctuation structures depend on the shape of the ITB. A low frequency (< 1 kHz) fluctuation with long distance correlation is observed in the “concave shaped ITB” plasma. After the spontaneous transition to the “weak curvature ITB”, a fluctuation at 6kHz, which has long correlation inside the ITB region, appears. Evolution of these fluctuations during transition indicates a strong correlation between the low frequency (< 1 kHz) fluctuation structure and the formation of the concave shaped ITB. These fluctuations with long radial correlation length also provide a clue to understand nonlocal response in transition.

EXC/8-1

Optimization of Density and Radiated Power Evolution Control using Magnetic ELM Pace-making in NSTX

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Recent experiments at the National Spherical Torus Experiment (NSTX) have shown that lithium coating of the plasma facing components leads to improved energy confinement, and also the complete suppression of edge-localized modes (ELMs). Due to the lack of ELMs, however, such plasmas suffer from density and radiated power that increase throughout the discharge, often leading to a radiative collapse. Previous experiments have shown that ELMs can be controllably restored into these lithium-conditioned discharges using 3D magnetic perturbations, which reduces impurity accumulation. Here we present the optimization of the use of magnetic ELM pace-making to control the evolution of the density and impurity content. Short duration large amplitude 3D field pulses are used, so that the threshold field for destabilization is reached and ELMs triggered quickly, and the field then removed. A second improvement was made by adding a negative-going pulse to each of the triggering pulses to counteract the vessel eddy currents and reduce time-averaged rotation braking. With these improvements to the triggering waveform, the frequency of the triggered ELMs was increased to over 60 Hz, reducing the average ELM size. The optimum frequency for attaining impurity control while minimizing energy confinement reduction was determined: fairly low frequency ELMs (20 Hz triggering) are sufficient to keep the total radiation under 1 MW throughout the discharge and avoid radiative collapse, with little reduction in the plasma stored energy. When combined with improved particle fueling the ELM-pacing technique has been successful in achieving stationary conditions in the line-averaged electron density and total radiated power, although the profiles continue to evolve.

EXC/8-4

Effect of ELM Mitigation on Confinement and Divertor Heat Loads on JET

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In JET several techniques have successfully demonstrated their capability for ELM amelioration (reduction of ELM losses and increase of ELM frequency), including resonant perturbations of the edge magnetic field by using the error field correction coils (EFCCs) and ELM magnetic triggering by fast vertical movements of the plasma column (vertical kicks). In this paper we present a summary of recent dedicated experiments in JET focused on integrating kicks and EFCCs into similar plasma scenario and scans in order to compare their performance. The impact of each control method on the plasma confinement and the effect of the reduction in ELM size on the divertor heat loads have been studied. In addition, the potential of using EFCCs for ELM mitigation in He^4 plasmas has also been explored in JET, and a comparison of those results with the effects observed in D plasmas is included in this paper. The plasma response to the application of these two ELM control methods in D plasmas shares common features: the reduction in ELM size is accompanied by a reduction in pedestal pressure (mainly due to a loss in density), resulting in a 10% reduction of the thermal stored energy. A key ingredient in the ELM mitigation experiments in JET is the diagnosis of the ELM-resolved divertor heat load profiles on the outer target by a fast high resolution infrared camera. It was found that the averaged ELM peak-power decreases almost linearly with the ELM size, but a smaller reduction is observed on the peak heat flux (40% for EFCCs when the reduction in peak power is $\sim 50\%$). This difference is related to a reduction in the width of the ELM heat flux profile. This observation is common to any small ELM regime, independently of the method employed to reduce the ELMs size (gas fuelling, kicks or EFCCs and also enhanced toroidal ripple). Interestingly, in the case of kicks or EFCCs the reduction in ELM losses is accompanied by some pedestal density loss, resulting in low pedestal collisionality. In contrast, a stronger reduction in the peak heat flux is observed in gas fuelled plasmas, where the reduction in ELM losses is correlated with an increase in collisionality. A detailed analysis of the divertor heat loads for mitigated ELMs will be compared to the analysis done for the database of spontaneous ELMs in JET. The implications of these new data for ITER will be discussed.

EXC/9-1

Impurity Transport of Ion ITB Plasmas on LHD

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An extremely hollow profile of carbon is observed associated with increase of ion temperature gradient in the Large Helical Device (LHD). 1) Transport analysis shows low diffusion coefficients of both carbon and bulk ions in the plasma core, and small outward convection of bulk ions and much larger outward convection of the carbon impurity. 2) The outward convection of carbon is considered to be due to the ion temperature gradient and it is opposite to the neoclassical prediction. 3) Difference of convection velocity in the magnetic axis position is clearly observed. It is suggested that the impurity hole becomes strong as the magnetic axis is shifted outward. 4) The dependence of the impurity hole on the atomic number, Z, is observed and the profile of higher Z impurity becomes more hollowed.

EXC/9-2

A Key to Improved Ion Core Confinement in JET Tokamak: Ion Stiffness Mitigation due to Combined Plasma Rotation and Low Magnetic Shear

Mantica, P.[1], Baiocchi, B.[1,2], Challis, C.[3], Johnson, T.[4], Salmi, A.[5], Strintzi, D.[6], Tala, T.[7], Tsalas, M.[6], Versloot, T.[8], De Vries, P.[8], Baruzzo, M.[9], Beurskens, M.[3], Beyer, P.[10], Bizarro, J.[11], Buratti, P.[12], Citrin, J.[8], Crisanti, F.[12], Garbet, X.[13], Giroud, C.[3], Hawkes, N.[3], Hobirk, J.[14], Hogeweij, G.[8], Imbeaux, F.[13], Joffrin, E.[13], Lerche, E.[15], McDonald, D.[3], Naulin, V.[16], Peeters, A.G.[17], Sarazin, Y.[13], Sozzi, C.[1], Van Eester, D.[15], Weiland, J.[18], and JET–EFDA Contributors[19]

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Detailed experimental investigation has been carried out on JET of the link between ITG driven ion stiffness and rotation and q profile. Rotating plasmas exhibit a reduced level of ion stiffness in the core region where q is flat, whilst they are very stiff irrespective of rotation in the outer region where q is steep. This was observed both using ion heat flux scans and T_i modulation and has led to a novel empirical hypothesis, that ion stiffness is mitigated by rotational shear in regions of low magnetic shear. Such hypothesis, although presently not supported by theory, has been confirmed by performing q profile scans at low and high rotation, with associated ion heat flux scan and T_i modulation to determine ITG threshold and stiffness. In the central region in low rotation plasmas the ions appeared very stiff and bound to threshold, with a beneficial effect of peaking the q profile, due to increased threshold as theory predicts; however at high rotation, the R/L_{T_i} was 3 times above threshold, following the marked reduction in stiffness level, and a deteriorating effect of peaking the q profile was observed, with the highest R/L_{T_i} observed for flat q profiles. In the outer regions ions appeared always very stiff and close to threshold. These observations may be at the basis of all situations of improved ion core confinement, such

as in Hybrid and ITB plasmas. Detailed investigation of hybrid plasmas revealed the existence of a core region of low ion stiffness, which is broader than in H modes due to the peculiar shape of the hybrid q profile. Such core improvement takes place in addition to the improvement of global confinement due to enhanced pedestal and absence of NTM. In ITB plasmas the core improvement becomes the dominant effect. A consequence of these results is that rotation is a necessary condition for achieving core ion improved confinement, which questions the feasibility of AT scenarios on ITER, unless a significant rotational shear can be induced.

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EXC/9-3

Long-Range Correlations and Edge Transport Bifurcation in Fusion Plasmas

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In order to understand the mechanism governing the development of transport barrier bifurcation, a European transport project has been recently promoted for studying the link between radial electric fields, long-range correlation (LRC) and zonal flows, and edge bifurcations in fusion plasmas. The main results are: (i) discovery of LRC and zonal flows in potential fluctuations that are amplified during the development of radial electric fields using electrode biasing in stellarators [TJ-II, TJ-K] and tokamaks [TEXTOR, ISTTOK] and during the spontaneous development of edge sheared flows in the TJ-II stellarator; (ii) comparative studies in tokamaks, stellarators and reversed field pinches have revealed significant differences in the level of the LRC, suggesting influence of magnetic perturbations in the long-range correlation; (iii) the degree of the LRC is strongly reduced as approaching the plasma density limit in the TEXTOR tokamak, indicating the possible role of collisionality and the impact of mean $\mathbf{E_r} \times \mathbf{B}$ flow shear as well on zonal flows.

EXC/9-4Rb

High-Beta Plasma Confinement and Inward Particle Diffusion in the Magnetospheric Device RT-1

Saitoh, H.[1], Yoshida, Z.[1], Morikawa, J.[1], Yano, Y.[1], Mizushima, T.[1], Ogawa, Y.[1], Furukawa, M.[1], Harima, K.[1], Kawazura, Y.[1], Tadachi, K.[1], Emoto, S.[1], Kobayashi, M.[1], Sugiura, T.[1], and Vogel, G.[1]

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Ring Trap 1 (RT-1) is a magnetospheric device that confines plasma in a dipole field generated by a levitated superconducting coil. An ultra high-beta state (possibly $\beta > 1$) of Alfvenic flowing plasma is theoretically predicted, where the plasma pressure is primarily balanced by the dynamic pressure of the fast flow. The final goal of RT-1 is to generate high-beta flowing plasma in the magnetospheric configuration capable of burning advanced fusion fuels. We have realized stable confinement of high-beta hot-electron plasma generated with electron cyclotron resonance heating (ECH) in RT-1. Geomagnetic field compensation and optimized operation have realized drastic improvements of the plasma properties, and the maximum local beta value has reached 70%. The maximum energy confinement time estimated from the diamagnetic measurements of plasma pressure and input RF power is 80 ms. Grad-Shafranov equilibrium calculation indicates that the pressure profiles have rather steep gradient near the superconducting coil in the strong field region. Cross field particle diffusion into the strong field region has been confirmed by pure electron plasma experiment. The spatial profiles of hot electrons were observed using a soft x-ray CCD camera. When the coil is not levitated, the major loss channel of the high temperature electrons is the coil support structure, from where we observed strong x-ray radiation.

EXC/P2 — Poster Session

EXC/P2-01

Performance Predictions of RF Heated Plasma in EAST

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The feature of high power L- and H-mode plasma in Experimental Advanced Superconducting Tokamak (EAST) that probably can be achieved in the spring campaign of 2010 is reported here. The simulation firstly uses PTRANSP code in combination with TSC code to explore the feature of EAST plasma with various radio frequency (RF) auxiliary heating methods, including ion cyclotron resonant heating (ICRH) system and lower hybrid current drive (LHCD) system. GLF23 transport model was found to be the best fit of experiment data in comparison with MMM95 and RLW-M models. Within the auxiliary power limits of the coming campaign, the electron and ion temperature in EAST plasma center can reach 2.2 and 2.8 keV respectively in H-mode plasma. The H factor and β_N will be 1.18 and 0.91. Possible discharge length of high power plasma can be $8.5 - 15$ s, depending on the volt-second consumption in different scenarios. Various phenomena are also reported, including the influence of different fraction of RF power on their deposition behavior, as well as on thermal diffusivity, the linear relation between q_0 and LHW power fraction, different behavior of fast ion between L- and H-mode plasma. The predicted feature of plasma can help physicists and operators design proposals. This study will be compared with experiments of the coming campaign to validate existing transport models in PTRANSP.

EXC/P2-02

Experiments and Simulations of ITER-like Plasmas in Alcator C-Mod

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Alcator C-Mod is performing ITER-like experiments to benchmark and verify projections to 15 MA ELMy H-mode Inductive ITER discharges. The main focus has been on the transient ramp phases. The plasma startup phase has been developed to include a large bore plasma rapidly after breakdown and to divert at about 80 ms out of a 500 ms current rampup time, close to the 15 s diverting time out of a 100 s current rampup planned for ITER. For most of the experiments presented, the plasma current in C-Mod is 1.3 MA and toroidal field is 5.4 T. Both ohmic and ion cyclotron (ICRF) heated discharges are examined. Plasma current rampup experiments have demonstrated that (ICRF) heating in the rise phase can save volt-seconds (V-s), as was predicted for ITER by simulations, and showed that the ICRF had no effect on the current profile versus ohmic discharges. The rampdown phase examined in experiments on C-Mod have durations of 350, 600, and 750 ms, equivalent to ITER rampdown times of 70, 120, and 180 s, respectively. The plasma elongation is reduced during the current rampdown from 1.8 to 1.4 to avoid vertical instabilities, and the plasmas remain diverted to 10% of their flattop plasma current. Rampdown experiments show an over-current in the ohmic coil (OH) at the H to L transition, as was predicted in simulations of ITER rampdown when the H to L transition occurred. This can be mitigated by remaining in H-mode into the rampdown. Experiments have shown that when the EDA H-mode is preserved well into the rampdown phase the density and temperature pedestal heights decrease during the plasma current rampdown. The delay of rise in li, afforded by remaining in H-mode, reaches over 200 ms early in the rampdown. Simulations of the C-Mod discharges have been done with the Tokamak Simulation Code (TSC) and the Coppi-Tang energy transport model is used both with modified settings to provide the best fit to the experimental electron temperature profile. The original model settings produced a more peaked temperature profile than observed, which made the simulated plasma li value too high in the L-mode rampup phase of the discharge. The impurities and associated radiation was treated by enforcing the measured total radiation profile and Z_{eff} from the experiment. The Bohm/gyro-Bohm and the CDBM energy transport models are also being examined for C-Mod ITER-like discharges.

EXC/P2-03

Solenoid-Free Startup Experiments in DIII-D

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A series of DIII-D experiments was performed to investigate the potential for initiating plasma current using poloidal field (PF) coils located outside the central solenoid region of the DIII-D tokamak, i.e. "solenoid-less startup" (SS). Inductive plasma current startup was achieved using only divertor and outer PF coils and strong electron cyclotron (EC) heating. Synergistic experiments were carried out using a standard solenoid generated plasma that transitions to non-inductive current drive (CD) using: EC, neutral beam (NB), and fast wave (FW). Conditions obtained in these latter experiments allowed a wide range of parameters to be studied than were obtained in SS experiments, and were designed to quantify the present DIII-D CD capabilities in low current, wall-limited plasmas. The key goal is to transition from the SS scenario to a fully noninductive CD scenario and to establish the feasibility of plasma initiation in a solenoidless tokamak, a requirement for viability of the low aspect ratio (LAR) path to fusion energy. Plasma current to 170 kA was achieved in the SS configuration limited by coil and power supply constraints in the DIII-D system. This was achieved utilizing ~ 2 MW of EC pre-ionization and heating. Flux conversion to plasma current was only slightly degraded relative to standard DIII-D startup. In the SS scenario, ECCD levels were small and prior to gaining closed loop radial position control, NBCD efficiency was decreased by plasma radial motion. Experiments using our standard solenoid initiation followed by noninductive CD provided valuable insight into the present DIII-D capabilities in plasmas typical of the SS configuration. While full noninductive current drive has been achieved in low current (340 kA), diverted, H-mode helium plasmas [1], the present DIII-D capability in a limited, L-mode plasmas has not been previously evaluated. Preliminary CD results indicate substantial noninductive current can be driven in a limited, L-mode plasma; however, self-sustaining levels with our present 6 co-injection system will require further optimization and possibly H-mode operation.

[1] T.C. Simonen, Phy. Rev. Lett. **61**, 15 1988

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EXC/P2-04

Demonstration of Multipulsed Current Drive Scenario using Coaxial Helicity Injection in the HIST Spherical Torus Plasmas

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The Helicity Injected Spherical Torus (HIST) device has been developed towards high-current start up and sustainment by Mutipulsed Coaxial Helicity Injection (M-CHI) method. Multiple pulses operation of the coaxial plasma gun can build the magnetic field of spherical torus (ST) and spheromak plasmas in a stepwise manner. Successive double gun pulses have been demonstrated to amplify the magnetic field and the plasma current against resistive decay. The resistive 3D-MHD numerical simulation has reproduced the current amplification by the M-CHI method and confirmed that stochastic magnetic field is reduced so that closed flux surfaces are created during the current drive. These experimental and computational results from STs have provided, for the first time, availability of a quasi-steady-state "refluxing" mode in which the magnetic field is allowed to decay partially before being rebuilt. Our goal is to achieve simultaneously the good energy confinement and the current sustainment by the M-CHI method.

EXC/P2-05

Experiment and Modeling of ITER Demonstration Discharges in the DIII-D Tokamak

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DIII-D is providing experimental evaluation of 4 leading ITER operational scenarios: the baseline scenario in ELMing H-mode, the advanced inductive scenario, the hybrid scenario, and the steady state scenario. The anticipated ITER shape, aspect ratio and value of I/aB were reproduced, with the size reduced by a factor of 3.7, while matching key performance targets for β_N and H_{98}. Since 2008, substantial experimental progress was made to improve the match to other expected ITER parameters for the baseline scenario. A lower density baseline discharge was developed with improved stationarity and density control to match the expected ITER edge pedestal collisionality ($\nu_e^* \sim 0.1$). Target values for β_N and H_{98} were maintained at lower collisionality (lower density) operation without loss in fusion performance but with significant change in ELM characteristics. The effects

of lower plasma rotation were investigated by adding counter-neutral beam power, resulting in only a modest reduction in confinement. Robust preemptive stabilization of 2/1 NTMs was demonstrated for the first time using ECCD under ITER-like conditions. Data from these experiments were used extensively to test and develop theory and modeling for realistic ITER projection and for further development of its optimum scenarios in DIII-D. Theory-based modeling of core transport (TGLF) with an edge pedestal boundary condition provided by the EPED1 model reproduces T_e and T_i profiles reasonably well for the 4 ITER scenarios developed in DIII-D. Modeling of the baseline scenario for low and high rotation discharges indicates that a modest performance increase of $\sim 15\%$ is needed to compensate for the expected lower rotation of ITER. Modeling of the steady-state scenario reproduces a strong dependence of confinement, stability, and noninductive fraction (f_{NI}) on q_{95}, as found in the experimental I_p scan, indicating that optimization of the q profile is critical to simultaneously achieving the $f_{NI} = 1$ and $Q = 5$ goals. Extended integrated modeling is being developed to improve capability for ITER projection by including the experimental observations of density peaking, ELM characteristics, NTM suppression and coupled core-edge-SOL transport.

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EXC/P2-06

Understanding Confinement in Advanced Inductive Scenario Plasmas – Dependence on Gyroradius and Rotation

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Hybrid and advanced inductive (AI) scenario plasmas have the potential for long pulse tokamak operation and high fusion yield. We have extended development of these plasmas to addressing issues important to extrapolation of plasma performance to ITER and beyond. First, we have carried out a series of experiments using JET and DIII-D to develop the scaling of confinement and transport with effective plasma size (ρ^* scaling with other dimensionless parameters fixed). A preliminary result of 0-D analysis is that the global scaling is close to Bohm-like: $B\tau_E \propto \rho^{*-(2+\alpha)}$ with α in the range $0.07 - 0.24$. Also, for well-matched discharges the confinement multiplier, H_{98y2}, depends weakly if at all on ρ^*, instead of the strong dependence seen in the ITPA database of hybrid plasmas [1]. 1D analysis of the profiles obtained for matched plasmas will yield the ρ^* scaling for ion and electron thermal diffusivities. Present experiments differ from ITER and future tokamaks in the magnitude of the expected plasma rotation and in the electron-ion temperature ratio. The dependence of confinement on rotation and on the presence of a neoclassical tearing mode island was determined in a DIII-D experiment in which the applied NB torque was varied in discharges with similar density and β, but for a range of q_{95} $(3.1 - 4.9)$. From the lowest accessible rotation to the highest, a range of ~ 4.6, H_{98} increased from ~ 2.0 to ~ 2.5, with a weak dependence on q_{95}. The dominant effect is the increase in $\mathbf{E} \times \mathbf{B}$ flow shear, with an accompanying decrease in low- and intermediate-k turbulence. The effect of decreasing 3/2 island width, while less important, is not negligible. Experiments at DIII-D have also addressed the dependence of confinement on T_e/T_i. Adding ECH to a hybrid scenario plasma increased T_e/T_i but also increased energy and momentum transport. Comparing plasmas matched in β and rotation but heated with co-NBI plus ECH vs co- plus counter-NBI showed that the reduction in density often associated with ECH is a consequence of reduced rotation rather than a change in T_e/T_i.

[1] D.C. McDonald, et al., Plasma Phys. Control. Fusion **50** (2008) 124013

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EXC/P2-07

Confinement of "Improved H-Modes" in the All-Tungsten ASDEX Upgrade

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"Improved H-mode" discharges in ASDEX Upgrade (AUG) are characterized by enhanced confinement factors $H_{98} > 1$, $\beta_N = 2 - 3.5$ and a q-profile with almost zero shear in the core of the plasma at $q(0) \sim 1$. One of the major goals of the AUG tungsten programme has been to demonstrate the compatibility of such high performance scenarios with an all-W wall. After the all-W AUG was boronised a clear reduction of the concentration of light impurities such as carbon and oxygen (C: $0.1 - 1\%$, $O < 0.1\%$) was observed. The radiated power decreased, especially in the divertor, and the thermal load on the W-coated divertor tiles reached values above technical capabilities. Therefore, high performance discharges in the boronized AUG were only conducted with active cooling of the divertor plasma by enhancing the radiation with N seeding. As a positive surprise it turned out that N seeding does not only protect the divertor tiles, but also improves significantly the energy confinement. This is a reproducible effect which holds for all D fuelling rates under both freshly boronised and unboronised conditions. In contrast to earlier studies of improved confinement following impurity seeding, density peaking, which would be detrimental in an all-W device, can be excluded as a contributor. The main contribution is the increase in the plasma temperature both in the core and in the edge. Stability analyses of comparable discharges with and without N seeding using the GS2 and the GENE codes highlight the role of deuterium dilution in the reduction of the core ion heat transport due to the ITG mode, which is dominant under the experimental conditions. The reduced core heat transport, however, explains the experimentally observed total confinement improvement only to a certain extent. This paper will deal with the present status of AUG plasma operation of "improved H-Mode" scenarios at optimized performance with boronized and unboronized tungsten walls. It will focus on confinement improvement observed with the introduction of N for divertor tile protection by detailed comparison of discharges with and without N puffing. In addition, linear and nonlinear gyrokinetic simulations of the core heat transport will be presented, concentrating on the role of additional impurity species of different type and helping to set limits to the core contribution to the overall confinement improvement.

EXC/P2-08

ITER Ramp-up and Ramp-down Scenarios Studies in Helium and Deuterium Plasmas in JET

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Recently, attention is given to documenting the current ramp up and current ramp down phase of tokamak discharges in preparation for ITER. The bulk of the experimental data from JET and other tokamaks have been obtained in deuterium. These results contributed to design modifications to the ITER coil set and divertor. The latest JET experiments are aimed at verifying whether the results obtained for ITER scenario studies in deuterium are still applicable in helium discharges. These experiments in helium duplicate the conditions of previous deuterium experiments at 2.66 MA and 2.36 T, using helium neutral beam injection and argon frosting of the divertor cryo-pumps to provide effective pumping of helium. Results show that flux consumption for helium discharges during plasma initiation is higher compared to deuterium. For the current rise phase, matched ohmic discharges show little difference in the plasma inductance values ($l_i(3)$) at the end of the current rise phase. Even an ohmic current rise in helium at $\langle n_e \rangle / n_{GW} \sim 0.5$, produces a value of $l_i(3) \sim 0.92$ at the end of the current rise. Helium discharges do confirm that good control of the plasma inductance is obtained by using a full bore plasma shape with early X-point formation at 0.8MA (equivalent to 4.5MA in ITER). Additional heating can keep $l_i(3)$ low < 0.85) during the ramp and H-mode transitions during the current rise are possible in helium plasmas. For the current ramp down, both helium and deuterium discharges show that it is important to remain diverted and in H-mode during the early phase of the ramp down while reducing the plasmas elongation. For the current ramp down phase, deuterium and helium discharges show little difference in $l_i(3)$ evolution and the flux consumption up to the end of the ramp down is similar. The experiments in JET show that deuterium and helium discharges are comparable with respect to key requirements for ITER.

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EXC/P2-09

Plasma Shape Feedback Control on EAST

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EAST (Experimental Advanced Superconducting Tokamak) is the first Chinese full-superconductive machine with D-shaped divertor configuration. In last two EAST operation campaigns, the shape feedback control was realized for diverted plasma using ISOFLUX control method in EAST PCS (EAST Plasma Control System). In a typical divertor plasma discharge, the plasma shape is feedback controlled from circular shape to elongated and fully diverted double null shape using different control algorithms. The smooth shape transition is essential for volt-seconds saving and plasma disruption avoidance. In order to achieve such aim, there is a trajectory used in EAST PCS for switching the control methods. The implementation details and experiment results will be presented in this paper. Now, EAST has obtained stable double-null plasma for longer than 60 seconds with auxiliary heating and current drive by Low Hybrid Wave. In the new started 2010 EAST campaign, single-null shape feedback control will be applied and further experiment results will be reported.

EXC/P3 — Poster Session

EXC/P3-01

L-H Transition Experiments in the TJ-II Stellarator

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In the TJ-II stellarator, spontaneous Low to High (L-H) transitions are achieved under NBI heating conditions when operating with lithium coated walls. H-mode transitions reproduce common features found in other devices: i.e. an increase in plasma density and plasma energy content, a reduction in Ha signal, the development of steep density gradients, an increase in the negative radial electric field and radial electric field shear, and a reduction in the density fluctuation level. Besides the observation of a significant radial electric field, the main result reported in this paper is the observation of remarkable fluctuating radial electric fields during the development of edge bifurcations, thus providing a new guideline for understanding the trigger mechanisms of the L-H transition. The dynamics of the radial electric field and density fluctuations reveal an increase in the fluctuations of the radial electric field and radial electric field shear within the frequency range $1 - 10$ kHz. Both, the increase in the low frequency oscillating radial electric field and the reduction in the broadband density fluctuations happen simultaneously at the transition and are detected a few milliseconds before the radial electric field shear development. Those observations may be interpreted in terms of turbulence suppression by oscillating sheared flows (zonal flows). The specific characteristics of the stellarator TJ-II allow controlling the position of low order rational values within the rotational transform profile and, therefore, the study of how the magnetic topology affects the L-H transition. Rotational transform scans show that both, the confinement enhancement factor and the radial electric field shear, increase at the L-H transition by an amount that depends on the magnetic configuration: higher values are obtained in configurations with a low order rational close to the plasma edge. Pronounced oscillations in radial electric field are detected at the transition that are absent in configurations without low order rational surfaces. These oscillations represent local changes in the radial electric field, which may be induced by the rational surface facilitating the transition. Further results on L-H transition experiments expanding the rotational transform and NBI heating power ranges are also discussed.

EXC/P3-02

Integration of a Radiative Divertor for Heat Load Control into JET Operational Scenarios

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The ITER-like Wall (ILW) project in JET is replacing its Plasma Facing Components currently made of Carbon-Fibre Composite with bulk beryllium main-chamber limiters with a full tungsten divertor: the material selection for the DT phase in ITER. The first concern that arises for JET plasma operation is to reduce PFC steady-state power loads to keep within the engineering limits for the ILW. The second is to limit the impurity production from the new PFC to maintain plasma performance, in terms of dilution (Be) and core radiation (W) by reducing the target plasma temperature. Both issues can be addressed by increasing divertor recycling and radiation. The latest JET experiments investigate the achievements of these aims by fuelling and/or impurity seeding of Ne and N_2, and assesses the impact on plasma performance on three main standard scenarios the ELMy H-mode, the Steady-State scenario and the Hybrid scenario. In the case of the ELMy H-mode, it was found that the incident power on the outer target can be reduced from 25% of the 16 MW input power, to 11% by D_2 fuelling alone with an acceptable degradation of the confinement, $H_{98(y,2)} \sim 0.95$, and a decrease of Z_{eff} by 0.2 from the reference value of 1.7. The power reaching the outer target can be reduced further down to 5%. of the input power, with the additional seeding of Ne and N_2 for a modest confinement loss, $H_{98(y,2)} \sim 0.9$. The Z_{eff} is moderately increased by 0.3 and 0.2 for Ne and N_2 respectively. It was found that by D_2 fuelling alone, the target temperature does

not decrease below 30 eV, with additional Ne not below 20 eV, while in the N_2-seeded cases it can be reduced to $10 - 15$ eV at the highest fuelling and seeding rates. The results of the ELMy H-mode and AT-like case studies will be extrapolated to the future carbon-free machine with the help of interpretative EDGE2D-EIRENE. It will be assessed whether there is any possibility to operate with D_2 fuelling alone at the present rate of input power and whether there is a domain of satisfactory confinement at the level of N_2 seeding compatible with a W core concentration in particular for the use of the 34 MW upgraded neutral beam power.

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EXC/P3-03

Transport in-between Edge Localized Modes in the Pedestal of ASDEX Upgrade

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The high confinement mode, or H-mode is characterized by an edge transport barrier (ETB), where transport by turbulence is reduced. The good confinement properties of the ETB are lost by the periodic occurence of type I edge localized modes (ELMs). Open issues are still the exact determination of the transport coefficients of impurities and electrons and of the radial electric field in the narrow ETB in-between ELMs, and the question what triggers an ELM? It is found that the impurity transport in-between ELMs in experimentally determined impurity density and temperature profiles, comes down to neoclassical values. Especially, an increase of the determined inward pinch coefficient v with the charge of the impurity ions (He^{2+}, C^{6+}, Ne^{10+}, Ar^{16+}) is found experimentally, in agreement with neoclassical transport predictions. Profiles of the radial electric field E_r in-between ELMs are inferred by analysing the He II line radiation at the plasma edge. The first results are according to neoclassical expectations. With the dedicated edge diagnostics the time evolutions of electron density, temperature, and pressure profiles in the ETB can now be observed during a type I ELM cycle. During the ELM crash the electron density and temperature profiles flatten. The re-steepening of the profiles occurs on different time scales: The electron density profile reaches its pre-ELM shape already about 4 ms after the ELM crash. The temperature profile can steepen quickly and sometimes gradually in the first 2.5 ms after the ELM crash and then the gradient stays constant for about 2 ms. When the density gradient has reached its pre-ELM value, the electron temperature gradient rises within the next 2 ms to its pre-ELM value. No ELM appears then, but large scale fluctuations of the electron density and temperature are observed. It is discussed whether the redistribution of the current density sets the time for the next ELM. The electron heat diffusivity in the ETB is anomalously high. It is not correlated with the observed large scale fluctuations, but with scale lengths down to the collisionless skin depth.

EXC/P3-04

Edge Sheared Flows as a Source of Propagating Plasma Potential Events

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The mechanisms underlying the generation of plasma flows play a crucial role to control transport in magnetically confined plasmas. Recent experiments performed in the TJ-II stellarator have shown that long-range correlations in potential fluctuations are present during the development of the edge sheared flows and that these correlations are amplified either by externally imposed radial electric fields or when approaching the L-H confinement edge transition. Experiments in the Alcator C-Mod tokamak show that the toroidal rotation propagates inwards radially from the plasma edge after the transition from low to high confinement regimes (L-H transition) and the resulting core rotation was found to depend strongly on the edge magnetic configuration. Other experiments also have shown regimes with spontaneous or anomalous poloidal rotation of the plasma core.

The main focus of this paper is the discovery of the edge sheared flows as a source of propagating inwards and outwards plasma potential events in the presence of sheared flows in TJ-II.

The radial structure of plasma fluctuations has been investigated in the plasma boundary region of the TJ-II stellarator. The comparison of results obtained in ECRH heated low density plasmas (below the plasma density threshold to trigger the development of $\mathbf{E}\times\mathbf{B}$ sheared flows) and NBI heated plasmas (high density plasmas) shows the existence of plasma potential events propagating outwards in the scrape-off layer side of the edge shear layer location and propagating inwards in the edge region with an effective radial propagation is the order of $1-10$ km/s. The inwards propagating events appear in the presence of sheared flows developed at the plasma edge.

The radial inwards / outwards propagation of potential events might reflect the momentum re-distribution linked to the development of sheared flows in the plasma boundary region. Considering that potential events propagating radially outwards will be lost in the SOL region or / and interacting with the plasma-wall whereas those propagating radially inwards will remain confined, these findings suggest the development of a momentum source linked to the edge velocity shear layer.

EXC/P3-05

Pedestal Turbulence Dynamics in ELMing and ELM-Free H-Mode Plasmas

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Large edge pedestal in H-mode plasma is critical for achieving high fusion power in the future burning plasma such as ITER. However, currently pedestal turbulence transport has not been understood fully, which requires knowledge of underlying turbulence properties. Characterizing pedestal turbulence is therefore crucial to understanding pedestal height limiting instabilities, as well as to testing theoretical models predicting the pedestal height and width. A series of recent experiments from DIII-D tokamak has provided several interesting results on the pedestal turbulence properties: In Type I ELMing H-mode discharges two distinct bands of the long wavelength density fluctuations measured by beam emission spectroscopy (BES), likely representing different underlying instabilities, are observed to be modulated with ELM cycle and propagating in opposite poloidal directions, along with their spatial and toroidal field dependence. The density fluctuation power builds up rapidly after ELM crash at high toroidal field but more slowly at low toroidal field, similar as the pedestal pressure gradients. In high electron pedestal pressure ELM-free QH-mode discharges, a set of high frequency coherent modes ($100-250$ kHz) are observed, after the disappearance of the low-frequency, low-n, edge harmonic oscillation. The features of these modes are similar to those predicted for kinetic ballooning mode (KBM), i.e., high n-number ($n \sim 20$), high decorrelation rates (exceeding local $\mathbf{E} \times \mathbf{B}$ shear rates), and poloidal wavenumbers comparable to ITG instabilities. The KBM is predicted to limit pedestal pressure gradient and is a basis for the EPED1 model predicting pedestal height and width. During RMP-ELM-suppressed plasmas, broadband fluctuations are significantly enhanced above those in ELMing discharges during application of the RMP. These enhanced fluctuations, observed from the pedestal into $r/a \sim 0.4$, are coincident with enhanced particle transport and may be partially or wholly responsible for reduced density during these discharges.

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EXC/P3-06

Power Requirements for Superior H-Mode Confinement on Alcator C-Mod: Experiments in Support of ITER

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Power requirements for maintaining suitably high confinement (i.e., normalized energy confinement time $H_{98} \geq 1$) in H-mode and its relation to H-mode threshold power, P_{L-H}, are of critical importance to ITER. In order to better characterize these power requirements and to complement prior examinations on JET [1], recent experiments on the Alcator C-Mod tokamak have investigated H-mode properties, including the edge pedestal, edge relaxation mechanisms and global confinement, over a range of input powers near and above P_{L-H}. In addition, we have examined the compatibility of impurity seeding with high performance operation, and the influence of plasma radiation and its spatial distribution on performance. Experiments were performed at 5.4 T at ITER relevant densities, utilizing bulk metal plasma facing surfaces and ion cyclotron range of frequencies waves for auxiliary heating. Input power was scanned both in stationary enhanced D_α (EDA) H-modes with no large ELMs and in ELMy H-modes in order to relate the resulting pedestal and confinement to the amount of power flowing into the scrape-off layer, P_{SOL}, and also to the divertor targets. In both EDA and ELMy H-mode, energy confinement is generally good, with H_{98} near unity. As P_{SOL} is reduced to levels approaching that in L-mode, pedestal temperature diminishes significantly and normalized confinement time drops. By seeding with low-Z impurities such as Ne and N_2, high total radiated power fractions are possible ($P_{RAD}/P_{TOT} > 0.7$), along with substantial reductions in divertor heat flux, all while maintaining $H_{98} \sim 1$. When the power radiated from the confined vs. unconfined plasma is examined, pedestal and confinement properties are clearly seen to be an increasing function of P_{SOL}, helping to unify the results with those from unseeded H-modes. This provides increased confidence that the power flow across the separatrix is the correct physics basis for ITER extrapolation. The experiments also help constrain the fraction of input power that can be radiated effectively in the edge with no degradation of core confinement, an item of key importance for ITER H-mode operation [2].

[1] Sartori, Plasma Phys. Control. Fusion Vol 46 (2004) 723.

[2] Loarte, Nucl. Fusion Vol 47 (2007) S203.

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EXC/P4 — Poster Session

EXC/P4-01

ITB Formation and MHD Activity in Experiments with Rational Surface Density Control

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Previous tokamak experiments show that ITBs are formed in the regions near the rational surfaces with low m and n numbers, where the rational surfaces density is very low, (the upper number of rational surfaces is supposed to be limited). If this interpretation is valid, we can organize ITB by external impacts in the plasma region, which we prefer. In T-10 experiments for this aim we used off-axis ECRH to suppress sawteeth and then rapidly ramp the plasma current with the rate up to 3 MA/s. The ITB appears at $q = 1.5$ surface after ECRH beginning and increases after the current ramp up. The ASTRA code calculation shows that $q(r)$ profile becomes very flat in plasma core. Near $q = 1.5$, the wide "gap"– the region without rational surfaces appears. The total energy confinement increase in the plasma core was up to a factor of 2.5. In some cases, when ITB reaches a high level, the internal disruptions at $q = 1.5$ appear. They decrease the ITB, but do not destroy it (that resembles the ELMy H-mode). Outside of ITB region, the self-consistent pressure profile exists. The link of this profile with sawteeth relaxations and other, more powerful MHD activity is discussed.

EXC/P4-02

Fluctuation-Induced Momentum Transport and Plasma Flows in the MST Reversed Field Pinch

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Fluctuations have been long considered a likely candidate for explaining energy, particle, and momentum transport in various magnetic confinement configurations. In particular, the physics of intrinsic plasma rotation, flow change associated with resonant magnetic perturbations, and momentum transport is of great importance to fusion plasma research as flows can act to improve performance. Plasmas in the Madison Symmetric Torus (MST) reversed-field pinch (RFP) rotate spontaneously and this rotation abruptly changes during repetitive bursts of current-driven MHD instabilities that stochastize the magnetic field and cause reconnection. In this paper, we report the first measurements of magnetic fluctuation-induced momentum flux in the core of the MST RFP, particularly during the sawtooth crash when the stochastic fields resulting from tearing reconnection are strongest. Direct measurements of the stochastic field driven momentum flux are made using a high-speed, laser-based, differential interferometer and Faraday rotation system. Measurements show (1) ion momentum flux from stochastic magnetic field can account for the equilibrium flow change at the magnetic axis and plasma edge during reconnection, and (2) multiple, global reconnections are required for momentum transport. When only the core resonant modes are excited and the edge modes remain small, the change in the plasma rotation and magnetic fluctuation-induced flux are largely eliminated. Ion momentum flux primarily arises from finite radial magnetic field correlation with density fluctuations. The origin of these density fluctuations is identified and related to nonlinear three-wave interactions.

EXC/P4-03

New Experiments on the GOL-3 Multiple Mirror Trap

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GOL-3 is a linear system intended for dense plasma confinement in a multiple-mirror magnetic field. The 12-m-long trap consists of 55 elementary mirror cells with 4.8/3.2 T magnetic field. The plasma density was $2 \times 10^{20} - 5 \times 10^{21}$ m^{-3} at typical electron and ion temperatures up to $1 - 2$ keV. The plasma is heated by a high-power electron beam ~ 0.8 MeV, ~ 20 kA, $\sim 12\,\mu$s, ~ 120 kJ).

Recent GOL-3 experiments were aimed at studies of the transverse transport. In the essentially new regime the heating of the deuterium plasma was done by a reduced-cross-section electron beam with the current decreased tenfold at the current density of ~ 1 kA/cm^2 (the same as in the “full-scale” experiments). Comparative analysis of shots into a plasma or neutral deuterium or in vacuum was done. The efficiency of collective relaxation of the beam reaches $\sim 50\%$ at maintained confinement time (~ 1 ms) despite the decreased beam cross-section.

A sub-THz plasma emission at and near the double plasma frequency ($0.25 - 0.4$ THz) was for the first time studied with a radiometric system based on multilayered selective elements and Shottky diodes. Such emission is a well-known marker of two-plasmon merger at non-linear Langmuir turbulence. The peak of emission spectra was observed at double plasma frequency and full duration of the emission corresponds to duration of the plasma heating. An irregular high-frequency structure of sub-THz emission with typical time scale of $2 - 10$ ns was observed.

Dynamic displacement of “thin” electron beam was observed in these experiments. Mode spectrum of perturbation of a magnetic surfaces was analyzed with new magnetic diagnostics. Stabilization of some MHD modes by a controlled injection of a heavy gas close to an exit receiver plate was demonstrated.

EXC/P4-04

Dependence of Particle Transport on Collisionality, Rotation and MHD in NSTX

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The NSTX spherical torus (ST) is a low-aspect ratio tokamak ($A < 1.5$) that is able to sustain a high plasma β operating with $B_\phi \sim 0.35 - 0.55$ T, $I_p \sim 0.7 - 1.1$ MA and neutral beam injection (NBI) heating power up to 7.0 MW. One of the ST predicted benefits is the reduction of the anomalous ion transport resulting in low core ($r/a < 0.7$) particle diffusivities in good agreement with the values predicted by neoclassical transport theory. Particle and impurity transport properties at low-aspect-ratio remain important for extrapolation to future ST-based devices such as an NHTX and CTF as well as to conventional aspect ratio schemes such as ITER. This paper will describe the results obtained from H-mode experiments aimed at studying, a) impurity transport in a $\nu^* \sim 1/T^2$ scan of NBI heated H-mode plasmas, b) the relationship between Pfirsch-Schluter particle transport and toroidal rotation and, as a by-product of a strong impurity seeding, c) the correlation between the strength of the impurity radiation emitted and the appearance of neoclassical tearing modes (NTMs).

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EXC/P4-05

Particle Deposition, Transport and Fuelling in Pellet Injection Experiments at JET

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Fuelling of reactor grade plasmas will become more critical than in existing Tokamaks due to high plasma temperatures that limit the penetration of neutral particles from the edge. Also, the penetration of energetic neutral beams is predicted to be rather shallow. Pellet injection is expected to become the routine tool for overcoming these problems. The viability and the efficiency of this method need to be further investigated in large tokamaks in order to gain confidence in the extrapolation to ITER. Pellet ablation and particle deposition have been extensively studied at JET giving a clear evidence of ∇B drift effects in hot plasmas and its impact on fuelling efficiency. This result confirms, for the first time on JET, observations made on smaller devices. In L-mode or moderately heated H-mode plasmas this effect seems to be very weak while at high temperature, injection from the Vertical High Field Side (VHFS) has proven to be much more effective than from the Low Field Side (LFS) in depositing particles beyond the pedestal. In the latter case ablated particles are quickly lost due to a combination of ∇B drift and edge instabilities. The fuelling performance, although investigated only in a limited range of plasma and pellet parameters, seems to be promising for the capability of pellets to raise the density without increasing the neutral pressure in the main chamber. This latter feature is important for ITER, because confirms that pellets are less demanding in terms of pumping capabilities. Experiments of pellet fuelling in combination with ELM mitigation techniques using Error Field Correction Coils (EFCC) will also be reported.

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EXC/P4-06

Experimental Study of Plasma Confinement on EAST

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Long pulse plasmas with double null divertor configuration have been successfully achieved by LHCD on EAST in 2009. The main parameters are: $I_p = 250$ kA, $B_t = 2.25$ T, $n_e = 1 \times 10^{19}$ m^{-3}, and plasma duration time is up to 60 s. The maximum injected power of LHCD is 1.2 MW at 2.45 GHz. The confinement of LHCD discharge is studied on EAST experimentally, it is in good agreement with the ITER-89 scaling law for L-mode plasma. It is observed that the highest energy confinement time is about 90 ms at plasma current of 0.25 MA during LHCD. The confinement of ohmic heating plasmas matched the neo-Alcator scaling law on EAST. The highest energy confinement time is about 200 ms in ohmic discharges. H-mode plasma is expected by the synergy of ICRF heating (4 MW at 30 – 110 MHz) and LHCD (1.5 MW at 2.45 GHz) on EAST in 2010.

EXC/P4-07

Model/Data Fusion: Developing Bayesian Inversion to constrain Equilibrium and Mode Structure

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Recently, Bayesian probability theory has been used at a number of experiments to fold uncertainties and interdependencies in the diagnostic data and forward models, together with prior knowledge of the state of the plasma, to increase accuracy of inferred physics variables. A new probabilistic framework, MINERVA [1], based on Bayesian graphical models, has been used at JET and W7-AS to yield predictions of internal magnetic structure. A feature of the framework is the Bayesian inversion for poloidal magnetic flux without the need for an explicit equilibrium assumption. We discuss results from a new project to develop Bayesian inversion tools that aim to (1) distinguish between competing equilibrium theories, which capture different physics, using the MAST spherical tokamak; and (2) test the predictions of MHD theory, particularly mode structure, using the H-1 Heliac.

Recent developments in MAST include the development of current tomography using Bayesian inversion of motional Stark effect data, the Bayesian inference of electron temperature and pressure from Thomson scattering data, and the inference of ion temperature and ion thermal flow from charge exchange recombination data. By estimating the thermal pressure profile from Thomson scattering data, combining this with toroidal current,

and assuming the plasma obeys the Grad-Shafranov equation, we have also computed the toroidal flux function, and corrected the toroidal field in MSE model to account for poloidal plasma currents. In new work, we present first results on the modelling and extraction of poloidal currents in MAST using Bayesian inference, and present first results on the development of an equilibrium verification framework to test different equilibrium physics models.

We also present new developments of Bayesian analysis to diagnostic systems in H-1, including electron temperate inference using multiple emission line measurements and a collisional radiative forward model. These developments enable the Bayesian inference of MHD mode structures, which have been computed by a reduced dimension cylindrical model as well as new results from fully 3D eigenmodes computed using CAS3D, a 3D ideal MHD stability code.

[1] J. Svensson, A. Werner, "Large Scale Bayesian Data Analysis for Nuclear Fusion Experiments", Proc. IEEE Workshop on Intelligent Signal Processing, 2007.

EXC/P4-08

Heat and Momentum Transport of Ion Internal Transport Barrier Plasmas on Large Helical Device

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The peaked ion-temperature profile with steep gradient so called ion internal transport barrier (ion ITB) was formed in the neutral beam heated plasmas on the Large Helical Device (LHD) and the high-ion-temperature regime of helical plasmas has been significantly extended. The ion thermal diffusivity in the ion ITB plasma decreases down to the neoclassical transport level. The heavy ion beam probe (HIBP) observed the smooth potential profile with negative radial electric field (ion root) in the core region where the ion thermal diffusivity decreases significantly. The large toroidal rotation was also observed in the ion ITB core and the transport of toroidal momentum was analyzed qualitatively. The decrease of momentum diffusivity with ion temperature increase was observed in the ion ITB core. The momentum transport driven by ion temperature gradient so called intrinsic rotation is also identified.

EXC/P4-09

ITB Formation During Slow Electron and Ion Heat Pulse Propagation in Tokamaks

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In T-10, ITB has been recently recognized by means of analysis of heat pulse propagation (HPP) induced by central ECRH-onset and cold pulse propagation (CPP) by off-axis ECRH cut-off in a sawtooth-free plasma created by off-axis ECRH. The cold pulses propagate slowly and diffusively with dynamic electron heat diffusivity $\chi_e^{HP} \sim 0.1$ m^2/s. It has been known for many years that in the L-mode, χ_e^{HP} values are few times greater than the power balance values χ_e^{PB} (the so called "enhanced" HPP). In the present report, we focus on the fully opposite cases with $\chi_{e,i}^{HP}$ much lower than χ_e^{PB} values ("reduced", or "slow" HPP). Non-local confinement bifurcations (jump of core transport at $\sim 0.3-0.4r/a$ region inside and around ITB in a ms timescale) were observed in JT-60U and T-10 plasmas and called the ITB-events. Slow outward electron and ion HPP ($\chi_{e,i}^{HP} \sim 0.1$ m^2/s) induced by ITB-event was observed in JT-60U. The new method allows us to approximately reconstruct behaviour of $\chi_{e,i}(r,t)$ during slow electron and ion HPP. In T-10, the reconstructed decay of χ_e correlates well with the value of χ_e at the end of CPP (obtained independently from the power balance). The fluctuations level measured by reflectometer falls below the Ohmic level in shots with slow CPP in T-10. In J-60U, the gradual decay of $\chi_{e,i}$ during HPP is accompanied by the rise of the radial electric field E_r shear calculated with neoclassical velocity of the poloidal rotation. In contrast to this, E_r just begins to vary slowly after ITB-event. The ITB splitting in two parts around the low-order rational values of q_{min} has been reported earlier in JET and JT60U. The localization of MHD-activity and diffusive inward HPP (induced by the ITB-event) across the zone between the two ITBs with $\chi_e^{HP} \sim 1$ m^2/s in JT60U (described in the present report) allows us to suppose the location of $q_{min} = 2.5$ position at the inner edge of the strong ITB. Therefore, the zone between the ITBs is located in the region with small negative shear, and the reason for ITB splitting is not the appearance of large magnetic island.

EXC/P4-10

Internal and Edge Electron Transport Barriers in the RFX-Mod Reversed Field Pinch

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In the Reversed Field Pinch RFX-mod electron Internal Transport Barriers (ITBs) with high electron temperature gradients are observed, intrinsically linked to the occurrence of a 3D magnetic equilibrium, featuring a helical core surrounded by an axisymmetric boundary. Electron External Transport Barriers (ETBs) develop as well, with ITBs or independently on them. In analogy with Tokamaks, ITBs are related to the magnetic

shear: a null point of the the q shear is found at the radial location of the internal barrier (q maximum). The flow profile could also play a role: 3D MHD simulations, supported by passive flow measurements, show a peak in the shear of the helical flow associated to the dynamo modes at q_{max}. In the region of the strong internal temperature gradients the thermal conductivity is reduced to $\sim$ 10 m^2/s, while increases inside the ITB, where the T_e profile flattens accordingly. Gyrokinetic calculations indicate that microtearing modes can contribute to the heat transport across the ITB more than ITGs, while investigations are presently in progress about the drive of transport inside the barrier. Experiments to fuel the central region by H pellet injection have shown that when the pellet penetrates the barrier, the density profile becomes transiently peaked. Inside the barrier the particle confinement is increased: the hydrogen diffusivity experimentally evaluated by reproducing the density profile evolution decreases to values around 5 m^2/s, of the same order than the neoclassical. Laser blow-off injection experiments have shown that, due to a strong outward convection around the ITB location, impurities do not penetrate the hot structure.

Strong edge T_e gradients are also observed, similar to pedestals. Such ETBs can be associated to a viscous effect, as they are correlated to the density (the higher edge T_e gradients are found at the lowest densities) and to a strong flow shear at the edge (indicating a turbulence damping). On the other hand, ETBs are also related to the magnetic topology: they develop at the innermost edge of the $m = 0$ island chain observed at the edge of RFX-mod, where the toroidal field reverses, and are more likely to be produced when the reversal surface is shifted towards the plasma edge with well conserved flux surfaces.

EXC/P4-11

Experimental Study of Potential Profile Formation in Large Helical Device

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The radial profile of potential in the core region of the Large Helical Device (LHD) was measured with a Heavy Ion Beam Probe (HIBP) in wide parameter area, in which radial electric field, E_r, could not be measured in previous experiments. The radial profiles of E_r were estimated from the differentiation of the experimental data of HIBP. These E_r profiles were compared with the calculation result from neoclassical theory, and the theoretical prediction almost coincided with the experimental data of E_r in the core region of LHD plasmas.

In toroidal magnetized plasmas, the radial electric field, E_r, is a very important parameter in transport phenomena. In the low collisional regime of helical devices, ripple induced diffusion is predicted from the neoclassical theory. This ripple-induced diffusion loss can be suppressed by the radial electric field, E_r, which is formed to satisfy the ambipolar condition. In previous experiments, charge exchange recombination spectroscopy (CXRS) was used to study E_r formation. However, most of data were obtained in the outer region ($r/a > 0.6$) of plasmas because of insufficient emission from the impurities in the core region. In order to clarify physics of E_r formation in the core region, where helical ripple is relatively small, the radial profile of plasma potential was measured with Heavy Ion Beam Probe (HIBP). In addition, we investigated E_r in wider plasma-parameter space including high ion temperature plasmas and high electron temperature plasmas heated by ECH (no NBI), in which the electric field measurements by CXRS is difficult. Experimental results are compared with calculation results from neoclassical theory, and we clarified that physics of neoclassical context is almost dominant in E_r formation in wide plasma parameters of advanced high temperature plasmas in LHD. These results will provide a physical basis to consider further improved confinement scenarios utilizing E_r in helical plasmas.

EXC/P4-12

Core Transport Properties in JT-60U and JET Identity Plasmas

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[9] *See Appendix of N. Oyama and the JT-60 Team, Nucl. Fusion* **49** *(2009) 104007.*
[10] *See Appendix of F. Romanelli et al., Nucl. Fusion* **49** *(2009) 104006.*

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Steady-state operation in ITER is foreseen to be with 100% non-inductive current drive at moderate plasma current ($I_p \sim 9$ MA) with $Q_{DT} \sim 5$ for a burning time of 3000 s. To compensate for the reduction of plasma current steady-state regimes requires the confinement to be improved beyond that usually achieved with an edge transport barrier, with $H_{IPB98(y,2)} \sim 1.5$. A possible solution consists in optimizing the core confinement with an Internal Transport Barrier, ITB. The purpose of the paper is to improve our understanding of the core confinement by comparing the transport properties in a set of identity experiments performed between JET and JT-60U in the advanced tokamak regime with ITBs. These transport physics ITPA-IEA joint experiments were performed with the same plasma shape, toroidal field (TF) ripple at the outer separatrix and normalized radial profiles: safety factor, normalized Larmor radius, normalized collision frequency, thermal beta, ratio of ion to electron temperatures etc. Similarities of the ITB triggering mechanism are observed when a proper match is achieved of the most relevant profiles of the normalized quantities. Similar thermal ion transport levels in the two devices are calculated in either monotonic or strongly reversed q-profiles. On the contrary, in reversed shear scenarios significant differences between JET and JT-60U are observed. We found that strongly reversed magnetic is required to trigger and sustain electron density ITBs. As a consequence of the peaked density profile, the core bootstrap current density is more than five times higher in JT-60U compared to JET. Thanks to the bootstrap effect, reversed magnetic shear configurations are self-sustained in JT-60U scenarios. Analyses of similarities and differences between the two devices address key questions on the validity of the usual assumptions made in ITER steady scenario modelling, e.g. a flat density profile in the core with thermal (ion/electron) transport barrier. Such assumptions have strong consequences on the prediction of fusion performance, bootstrap current and on the sustainement of the scenario.

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EXC/P4-13

Scaling of Density Peaking for Plasma with Pellet Injection

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This work aims to study the pellet ablation rate, the pellet penetration and the plasma behaviors during pellet injection in ITER plasma using 1.5D BALDUR integrated predictive modeling code together with the Neutral Gas Shielding (NGS) pellet ablation rate model. For this steady transonic-flow pellet ablation model, several extensions and modifications have been made over the years. In this work, the NGS pellet ablation model by Park and colleagues is used. The result shows that the density near the plasma edge increases but the temperature decreases significantly after the pellet injection is used. In addition, a scaling of electron plasma density peaking resulting from pellet is developed based on the simulation results.

EXC/P5 — Poster Session

EXC/P5-01

Helical Structures and Improved Confinement in the MST RFP

Chapman, B.E.[1], Almagri, A.F.[1], Bergerson, W.F.[2], Bonomo, F.[3], Brower, D.L.[2], Burke, D.R.[1], Clayton, D.J.[1], Den Hartog, D.J.[1], Ding, W.X.[2], Forest, C.B.[1], Franz, P.[3], Gobbin, M.[3], Goetz, J.A.[1], Hegna, C.C.[1], Kaufman, M.C.[1], Piovesan, P.[3], Reusch, J.A.[1], Sarff, J.S.[1], and Stephens, H.D.[1]

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There are two cases in MST where a central helical magnetic structure emerges and produces a region of improved confinement. One case is due to a single tearing mode dominating the core-resonant $m = 1$ mode spectrum. Here, runaway electrons are observed with energies > 100 keV [1], a feature normally absent in standard, stochastic RFP plasmas. These electrons are deduced to be confined in a locally non-stochastic region inside the dominant mode's magnetic island. The other case corresponds to an additional mode that emerges following global reconnection events. In this case, the $m = 1$ spectrum is fairly flat, but the electron temperature profile exhibits a local peaking corresponding to a substantially reduced electron thermal diffusivity [2]. While neither of these cases corresponds to a substantial improvement in global confinement, recent discharges near MST's maximum toroidal plasma current exhibit very peaked tearing mode spectra. These spectra bear a striking similarity to those in the RFX-mod RFP which produce a several-fold improvement in global confinement [3]. Helical structures in the RFP are of interest not only for their contribution to confinement improvement, but also for their connection to 3D physics in other configurations.

[1] D.J. Clayton, B.E. Chapman, R. O'Connell et al., Phys. Plasmas 17, 012505 (2010).

[2] H.D. Stephens, D.J. Den Hartog, C.C. Hegna et al., submitted to Phys. Plasmas.

[3] R. Lorenzini, D. Terranova, A. Alfier et al., Phys. Rev. Lett 101, 025005 (2008).

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EXC/P5-02

Transport and MHD Analysis of ELM Suppression in DIII-D Hybrid Plasmas using n = 3 Resonant Magnetic Perturbations

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Experiments performed at the DIII-D tokamak have successfully demonstrated complete suppression of edge localized modes (ELMs) in the hybrid scenario via the application of resonant magnetic perturbation (RMP). While high energy confinement ($H_{98y2} > 1$) is maintained, the interaction of a growing $m = 3$, $n = 2$ NTM causes the momentum confinement time to decrease by a factor of three. The plasma toroidal rotation slows, and below ~ 40 km s^{-1} ELMs are observed to return. These discharges, done in a shape similar to that planned for ITER, have pressure gradients in the edge that are believed to drive a parallel bootstrap current which is dominant over other current sources in this region. Steep pressure and current gradients can cause unstable MHD modes known as peeling-ballooning modes and are thought to drive large "Type-I" ELMs typical in the tokamak H-mode. Peeling-ballooning stability calculations, performed with the ELITE code, are dependent on both the edge pressure and current gradients and show that the ELM suppressed hybrid discharges are stable. The EPED1 code, which has successfully been used to predict the total pedestal height in standard ELMing H-mode discharges across a number of different machines, overestimates the total pedestal pressure height when applied to RMP discharges. Additionally, the inclusion of the edge bootstrap current has a non-negligible effect on the determination of magnetic field line stochasticity in the pedestal region; the driving mechanism behind RMP. Field line tracing and magnetic diffusion coefficients during RMP show a decrease of magnetic stochasticity when ELMs were not completely suppressed relative to a phase of complete ELM suppression.

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EXC/P5-03

Edge Plasma Behavior during Externally Applied Electrode Biasing and Resonant Magnetic Perturbation in IR-T1 tokamak

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Edge plasma behavior during externally applied electrode biasing and resonant helical magnetic field (RHF) perturbation in IR-T1 tokamak has been investigated. The experiments have been done in different regimes as electrode biasing, magnetic perturbation application and both of them. The profiles of radial electron temperature, floating potential, poloidal and toroidal rotation velocity have been measured by using of a movable Langmuir probe and Mach probes in the edge of plasma. Magnetohydrodynamic phenomena based on Mirnov oscillations have been investigated by applying the positive electrode biasing potential and RHF. An effect of RHF on magnetohydrodynamics behaviour during positive electrode biasing has been investigated. The results shown that subsequent to the application of a positive bias, an increase in the frequency of magnetic field fluctuations was observed but after RHF application during $L = 2$ and $n = 1$ (L poloidal and n toroidal turn) the $m = 2$ MHD mode oscillations amplified and $m = 3$ mode is suppressed; also by applying $L = 3$ RHF's the $m = 2$ MHD mode oscillation has been disappeared. The results analysed by SVD and wavelet methods. Simultaneous measurements of the radial electric field and poloidal magnetic field oscillations shown that magnetohydrodynamic activity is damped when the radial electric field becomes more negative in the plasma edge. The results shown that MHD frequency decreases while the radial electric field increasing negatively. The toroidal velocity changes after a short delay time of about $t_d = 1 - 1.5$ ms during RHF application while poloidal velocity changes just after RHF's. The poloidal velocity increased after positive bias but it decreased after RHF application during Bias regime. Toroidal velocity hasn't considerable change during biasing but it decreased after RHF application and poloidal Mach number decreased during biasing application too.

EXC/P7 — Poster Session

EXC/P7-01

Reduced Electron Thermal Transport in Low Collisionality H-mode Plasmas in DIII-D and the Importance of Small-Scale Turbulence

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Understanding electron thermal transport in tokamaks is of crucial importance in next generation burning plasma experiments where α-particles produced by fusion reactions primarily heat the electrons. The first systematic investigation of core electron thermal transport and the role of local ITG/TEM/ETG-scale core turbulence is performed in high temperature, low collisionality DIII-D H-mode plasmas. Core ITG/TEM-scale turbulence is substantially reduced/suppressed by $\mathbf{E} \times \mathbf{B}$ shear promptly after the L-H transition. As a result, a substantial reduction of the electron heat diffusivity across the entire minor radius within 10 ms of the L-H transition is found from time-dependent transport analysis. Initial nonlinear gyrokinetic (GYRO) simulations indicate that a significant portion ($> 50\%$) of the remaining H-mode electron heat flux results directly from residual short-scale TEM/ETG turbulence. The studies are performed at ITER-relevant collisonality ($\nu_e^* \sim 0.05$, $r/a \leq 0.6$) and are important since the ITER plasmas will be electron heat-dominated. Core turbulence wavenumber spectra, obtained via Doppler backscattering, indicate an exponential dependence of fluctuation levels on the normalized poloidal wavenumber $k_\theta \rho_s$ in L-mode. Substantially reduced ITG/TEM fluctuation amplitudes are found in H-mode within the wavenumber range ($0.4 \leq k_\theta \rho_s \leq 3$) where shear stabilization is expected from a comparison of linear instability growth rates and the flux-surface-averaged shearing rate. Taking advantage of the unique set of DIII-D turbulence and profile diagnostics, experimentally determined H-mode core turbulence spectra and transport fluxes are directly compared for the first time with nonlinear gyrokinetic simulation results. Initial GYRO calculations indicate flattened H-mode fluctuation spectra in the ITG/TEM spectral range consistent with measured wavenumber spectra. Multi-scale GYRO simulations with improved low-k resolution are underway to allow quantitative comparisons. The results presented provide evidence that ITG-scale density/electron

temperature fluctuations as well as intermediate-scale turbulence is significantly reduced in the core of high-performance H-mode plasmas, and that smaller-scale modes can play a substantial role in electron transport in these plasmas.

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EXC/P8 — Poster Session

EXC/P8-01

Non-Collisional Ion Heating and Magnetic Turbulence in the RFP

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Strong non-collisional ion heating occurs during sawtooth-like magnetic reconnection events in MST Reversed Field Pinch (RFP) plasma, where ions are transiently heated to as high as 3 keV, often exceeding the electron temperature. The source for this ion heating is the mean magnetic field energy. This transient ion heating has been utilized to enhance plasma beta and energy confinement in MST by timing the application of current profile control just after a reconnection event. The high frequency magnetic turbulence broadens and develops anisotropy with respect to the background magnetic field as expected in magnetized plasma. Magnetic fluctuations are measured over a broad range of scales down to the ion gyroradius. These fluctuations exhibit an exponential spectral density consistent with dissipative nonlinear turbulent cascade, originating from unstable MHD tearing fluctuations, believed to be related to the observed non-collisional ion heating (which is not understood). The non-collisional ion heating and anisotropic magnetic turbulence are also observed in astrophysical plasmas, heightening the interest in these observations. MST provides a laboratory setting to examine this ion heating mechanism and magnetic turbulence. Our key results are 1) achieved high ion temperature through a non-collisional heating and then sustained by parallel current profile control helping to establish MST's record energy confinement time of 12 ms, 2) the resulting ion temperature has a mass dependence where heavier ions heat up to higher temperature 3) the high frequency magnetic turbulence is anisotropic in wave number, and 4) we observe a dissipative cascade in kperp with $k_{dis} = 0.22$/cm.

EXC/P8-02

Vortex Confinement of Hot Ion Plasma with $\beta = 0.6$ in Axially Symmetric Magnetic Mirror

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A so called vortex confinement of plasma in axially symmetric mirror device magnetic field was studied. This recently developed approach enables to significantly reduce transverse particle and heat losses typically caused by MHD instabilities which can be excited in this case. Vortex confinement regime was established by application of different potentials to the radial plasma limiters and end-plates. As a result, the sheared plasma flow at periphery appears, which wraps the plasma core. Experiments were carried out on the gas dynamic trap (GDT) device, where hot ions with a mean energy of $E_h = 10$ keV and the maximum density of energetic ions $n_h = 5 \times 10^{19}$ m^{-3} were produced by oblique injection of deuterium or hydrogen neutral beams into a collisional warm plasma with the electron temperature up to 250 eV and density $n_w = 2 \times 10^{19}$ m^{-3}. Local plasma beta approaching 0.6 was measured. The measured transverse heat losses were considerably smaller than the axial ones. The measured axial losses were found to be in a good agreement with the results of numerical simulations. Recent experimental results support the concept of the neutron source based on the gas dynamic trap.

EXC/P8-03

Confinement and Edge Studies towards Low ρ^* and ν^* at JET

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High plasma current operation at JET allows access to pedestal parameters similar to those expected for the ITER baseline scenario at $Q_{DT} = 10$ ($T_{ped,ITER} \sim 3 - 5$ keV, $n_{e,ped,ITER} \sim 10^{20}$ m^{-3}) approaching the ITER dimensionless parameters ($\nu^*_{JET} \sim 0.05$ and $\rho^*_{JET} \sim 4.2 \times 10^{-3}$). Results from previous experiments at JET showed at high plasma currents (above 3.8 MA), a deviation of the energy confinement from the IPB98$_{(y,2)}$ scaling law. New diagnostic capabilities for pedestal measurements and high neutral beam heating power (24 MW), motivated JET to revisit the questions previously raised, in particular the effect of ν^* and beta normalised as well as of ρ^* on global confinement. The new experiments include a scan of beta normalised at constant collisionality for plasma currents up to 3.5 MA at fixed $q_{95} = 3$ with confinement enhancement factor of 1 and discharges with plasma currents up to 4.5 MA at $q_{95} = 2.67$ and $\beta_N \sim 1.4 - 1.5$. Comparing the experimental energy confinement obtained in the new experiments with the prediction from the scaling law it is found that for $I_p > 3.8$ MA the plasma stored energy is lower than the one predicted by the scaling law in line with previous results. Several reasons for the decrease of energy confinement at high I_p have been investigated: intrinsic confinement deterioration at low ρ^*, beta normalised dependence, proximity to the H-mode power threshold (PL-H), MHD and gas fuelling. Dedicated experiments showed only a weak or no variation of confinement with ρ^* relative to the scaling law. The dependence of confinement with beta normalised shows a positive correlation at high I_p; the normalised confinement is seen to decrease with decreasing beta normalised. The role of MHD stability and its influence on the pedestal pressure is investigated for the highest I_p where q_{95}, ν^* and beta normalised are varied at the same time as ρ^*. The proximity to PL-H has been previously observed to be associated with reduced confinement in previous studies. In these experiments, a comparison of the power above threshold between the low plasma current discharges and the high current discharges shows that the latter lies closer to the PL-H. The effect of gas fuelling on the normalised confinement and the impact of the ELM crash on the PFCs with ELM size in the range of 0.2 MJ to 0.9 MJ is also presented.

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EXC/P8-04

Plasma Rotation and Transport in MAST Spherical Tokamak

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The role of the toroidal rotation driven by the NBI heating in determining the confinement properties of MAST spherical tokamak plasmas is investigated. The resulting $\mathbf{E} \times \mathbf{B}$ flow shear can be strong enough to suppress low-k ion scale transport close to the neo-classical level in the plasma core, although the electron transport remains anomalous. High resolution kinetic- and q-profile diagnostics on MAST allow the conditions favoring the formation of internal transport barriers to be investigated, including the role of $\mathbf{E} \times \mathbf{B}$ flow and magnetic shear, MHD and rational surfaces. Poloidal rotation measured using CXRS is found to be small (a few km/s) and comparable in magnitude to the predictions of neo-classical theory, hence making a small contribution to flow shear in the plasma core. Formation of ITBs in L-mode plasmas, in which the ion transport is also highly anomalous in the outer regions, is favoured by conditions of high toroidal flow and reversed magnetic shear. With co-current directed NBI heating, ITBs in the momentum and ion thermal channels tend to form in the vicinity of q_{min}, particularly when this passes through a rational value. In contrast, with counter-NBI ITBs of greater radial extent form outside q_{min} due to the broader profile of $\mathbf{E} \times \mathbf{B}$ flow shear produced by the greater prompt fast-ion loss torque. A number of gyrokinetic or MHD fluid turbulence codes are now available which incorporate the physics of flow shear. A beam emission spectroscopy (BES) system for low-k turbulence measurements is to be installed in 2010 and a synthetic BES diagnostic has been developed which will allow direct comparison of the turbulence characteristics with results of the simulations.

EXC/P8-05

Local Transport Property of Reactor-Relevant High-Beta Plasmas on LHD

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High-beta plasmas of more than 5% of the volume averaged beta, which are required for an economic fusion reactor of helical type, were obtained on the Large Helical Device (LHD). No degradation seems to appear on the global confinement property, which is compared with the International Stellarator Scaling 2004 (ISS04), in such high-beta plasmas, when effects of the change of the magnetic configuration by the Shafranov shift due to the increment in beta are considered. In order to confirm the possibility of extrapolation of this global confinement property on LHD to reactor plasmas, it is studied whether the local transport property is preserved in the high-beta region. The local transport coefficients, χ^{eff}, are derived based on the power balance in the steady state. The coefficient χ^{eff} is normalized by a reference transport coefficient, χ^{ISS04}, which has the same non-dimensional parameter dependence as ISS04. At first, the dependence of $\chi^{\mathrm{eff}}/\chi^{\mathrm{ISS04}}$ on the magnetic configurations is studied. ISS04 contains the renormalization factor, f_{ren}. For the local transport coefficients, a new renormalization factor, $g_{\mathrm{ren}\chi}$, which represents the dependence on the magnetic configurations is introduced. From the evaluation of $g_{\mathrm{ren}\chi}$, it is found that the dependence of the local transport property on the magnetic configurations in the low-beta region is almost coincides with the property which is derived from the global confinement scaling of ISS04. Then, the beta dependence of the local transport coefficients is studied in the wide range of beta. The factor $g_{\mathrm{ren}\chi}$ is interpolated for the geometric center position of the magnetic flux surface. This interpolated parameter, $g_{\mathrm{ren}\chi}^{\mathrm{int}}$, represents the effects of the change of the magnetic configurations on the local transport coefficients in the high-beta region. The beta dependence of the ratio $\chi_{\mathrm{eff}}/(g_{\mathrm{ren}\chi}^{\mathrm{int}}\chi^{\mathrm{ISS04}})$ shows the effects other than the change of the magnetic configuration by the increment in beta. It is found that the local transport property, which is evaluated by referring to ISS04, is preserved or improved in the core region, while it is degraded in the peripheral region.

EXC/P8-06

Particle Transport in Vanishing Turbulence Conditions in the Tore Supra Plasma Core

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Maximisation of the fusion reaction rate in a future reactor will be achieved, among others, by maximising the fuel (and hence electron) density and minimising the impurity density in the plasma core. The path to this objective will depend on the physics process, collisions or turbulence, which will drive transport. This question is of particular interest in plasmas affected by the so-called sawtooth instability. Indeed, this will affect the ITER central plasma volume with a period much longer than the energy confinement time. Electron and impurity transport in Tore Supra plasmas affected by “ordinary” to monster sawtootheeth has been investigated. Besides the transition from neoclassical to turbulent behaviour, the study reveals an unexplained anomalous contribution to electron diffusion.

In the core of ohmic sawtoothing plasmas, the electron convection velocity is found to be inward and slightly stronger than the calculated Ware pinch within narrow error bars while diffusion, although lower than in the gradient zone, is an order of magnitude higher than the neoclassical prediction. In contrast with electrons, impurity transport in the same situation is unambiguously consistent with the neoclassical predictions, which is more consistent with the low level of measured density fluctuations (about 0.2%). Moreover, impurity diffusion is lower than the anomalous electron diffusion. Several phenomena, such as turbulence spreading or resonance broadening, are discussed.

The sawtooth regimes have been explored by injection of ICRH power up to 5 MW. The density peak amplitude, which reaches 10% of the central density in ohmic regime, decreases with increasing power and disappears at 4 MW in plasmas modulated by giant sawteeth. This indicates either an enhanced diffusion, in qualitative agreement with the very high impurity diffusion coefficient in this situation and the short measured gradient lengths, or a vanishing convection velocity. Neoclassical and turbulent transport models are used to interpret these observations.

EXC/P8-07

Electrode Biasing Experiment in the Large Helical Device

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In neoclassical theories the nonlinearity of the ion viscosity plays the important role in the bifurcation phenomena of the L-H transition, which observed in tokamaks and stellarators. The effects of the ion viscosity maxima on the transition to an improved confinement mode were experimentally investigated by the externally controlled $\mathbf{J} \times \mathbf{B}$ driving force for a poloidal rotation using the hot cathode biasing in Tohoku University Heliac (TU-Heliac) [1-3]. In steady state the $\mathbf{J} \times \mathbf{B}$ driving force balances with the ion viscous damping force and the friction to neutral particles. Then the ion viscosity opposing to the poloidal rotation can be estimated experimentally by subtracting the friction term from the driving force. Therefore it is important to verify whether the transition phenomena observed in TU-Heliac can appear in the wide plasma parameter range and in the confine systems which have different magnetic configuration. The importance of the biasing experiment in stellarators exists in the ability to understand universally the relation between the transition behavior and the viscosity by taking high order magnetic Fourier components into viscosity evaluation. The optimization of helical ripples allows the reduction of viscosity, which is expected to bring good accessibilities to the improved confinement modes.

In this paper we report the electrode biasing experiments in CHS and LHD. The transition to the improved confinement mode by the electrode biasing was observed for the first time in LHD. The negative resistance was observed in the confinement mode sustained by the cold electrode biasing. The electrode current showed the clear decrease against to the increase in the electrode voltage and had hysteresis in the transition phenomena. The decrease in the electrode current suggested the improvement of the radial particle transport. The increase in the energy confinement time and remarkable suppression of the density fluctuation correspond to the transition were also observed. These results indicated that the electrode biased plasma in LHD showed the similar improvements in confinement to the observations in the H-mode plasma in tokamaks and stellarators.

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EXC/P8-08

Transport and Turbulence with Innovative Plasma Shapes in the TCV Tokamak

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We present recent results on turbulence measurements in TCV L-mode plasmas. It has been shown that the heat transport is reduced by a factor two for a plasma at negative triangularity compared with a plasma at positive triangularity. This transport reduction is reflected in the reduction of the temperature fluctuation level, in the low frequency part of the spectrum ($30 - 150$ kHz), measured by correlation ECE in the outer equatorial plane. Moreover, for identical electron density and temperature profiles: the radial correlation length is typically reduced by a factor of two at negative triangularity. Nonlinear gyrokinetic simulations predict that the TEM turbulence might be dominant for these TCV plasmas. The TEM induced transport is shown to decrease with decreasing triangularity and increasing collisionality. Both dependences are in fairly good agreement with experimental observations. First results of an experimental scenario to reach the H-mode regime at negative triangularity in TCV are presented. We report on an innovative divertor magnetic configuration: the snowflake (SF) divertor whose properties are expected to affect the local heat load to the divertor plates in particular during ELMs compared to the classical single-null (SN) divertor. Snowflake diverted plasmas, in both L-mode and H-mode are routinely obtained in the TCV tokamak. For ELMy SF plasmas, the strike-point properties (density, temperature, fluctuation level, ...) are studied with Langmuir probes (LPs) and compared with those of a SN configuration obtained within the same discharge. In L-mode plasmas, the intermittent particle and heat transport in the SOL is associated with the presence of “blobs” propagating in the radial direction. Intermittency is compared between SN and SF configurations by looking at the statistical properties of ion saturation current fluctuations at the LFS wall.

EXC/P8-09

Turbulence and Transport in Simple Magnetized Toroidal Plasmas

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Progress in understanding turbulence and related cross-field transport of fusion relevance is achieved in the simple magnetized plasmas of the TORPEX toroidal device, which provides a test bed for code benchmarking and theory validation. In TORPEX ($R = 1$ m, $a = 0.2$ m), a small vertical magnetic field ($B_z < 4$ mT) is superposed on a toroidal magnetic field $B_t < 100$ mT to form open helicoidal magnetic field lines, which, similarly to Scrape-Off Layers of fusion devices, feature grad-B and magnetic field curvature. Plasmas of different gases are produced by microwaves (2.45 GHz, $P < 10$ kW) with typical electron temperatures $\sim 5 - 15$ eV and densities ranging from $\sim 3 \times 10^{16}$ m^{-3} for hydrogen to $\sim 3 \times 10^{17}$ m^{-3} for neon. For large B_z amplitudes, the ideal interchange instability is observed and characterized. The existence of a critical pressure gradient needed to drive the ideal interchange instability is experimentally demonstrated using a scan in the neutral gas pressure. The dynamics of density blobs born from ideal interchange waves is studied using a novel spatiotemporal pattern recognition method, allowing the determination of statistical observables. We investigate the scaling of the cross-field blob velocity with blob size, connection length and neutral gas pressure for different gases. An analytical expression for the blob velocity including cross-field ion polarization currents, parallel currents to the sheath, and ion-neutral collisions is derived and shows good quantitative agreement with the experimental data. We report on the link between toroidal flows and blobs: toroidal momentum is transferred from an ideal-interchange mode to density blobs. Depending on the phase between the toroidal flow and the density in the mode region, blob-induced flows are either dipolar structures or monopolar perturbations, so large that the toroidal flow gets transiently reversed. The turbulent toroidal momentum flux is dominated either by the nonlinear flux or by the convective part but not by the Reynolds stress component. We present also 3D global fluid nonlinear simulations, which reveal the existence of a resistive interchange instability at low B_z while the drift wave regime is almost inaccessible at realistic collisionality. The experimental validation of these predictions is under way.

EXC/P8-10

Ohmic and NBI Heating in the TUMAN-3M with Increased Toroidal Magnetic Field

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Anomalous electron transport is one of the main issues in physics of plasma heating in MCF devices. Besides fundamental interest the motivation of the study of toroidal magnetic field (B_t) effect on plasma heating and specifically on electron heating is the intention to increase efficiency of Neutral Beam Injection (NBI) in a low B_t tokamak. Recent upgrade of power supply in the TUMAN-3M allowed to increase B_t in the NBI phase from 0.68 to 1.0 T. In these conditions a study of B_t influence on the energy confinement time and electron temperature in ohmically and NBI heated H-mode plasmas was performed. Range of accessible plasma currents the increased B_t was extended from 140 to 190 kA. Thus, effect of plasma current on confinement was studied as well. Electron temperature T_e was deduced from measurements of the SXR emission, ion temperature T_i was measured by NPA analyzer and stored energy was obtained from diamagnetic data. Behavior of NBI born fast ions (FI) was analyzed using measurements of 2.45 MeV D-D neutron fluxes. In the ohmic H-mode T_e was observed to rise from $0.4 - 0.5$ to $0.55 - 0.6$ keV when B_t was increased from 0.68 to 1.0 T. Observed increases in T_e and stored energy Wdia indicate strong B_t dependence of the energy confinement time: $\tau_E \sim B_t^{0.75}$. The dependence is much stronger than $\tau_E \sim B_t^{0.15}$ predicted by IPB98$_{(y,2)}$ H-mode scaling. Increase in the plasma current from 140 kA to 190 kA at $B_t = 1.0$ T resulted in the further growth of T_e up to $0.65 - 0.75$ keV and clear increase in W_{dia}, which are in a good agreement with IPB98$_{(y,2)}$ prediction: $\tau_E \sim I_p^{0.93}$. Flux of 2.45 MeV D-D neutrons was found to rise by a factor of 2 in the conditions of increased B_t, thus indicating improvement in the capture and confinement of fast ions. In agreement with this observation the neutron flux decay time after NBI switch-off-tau-n was increased substantially. Lesser plasma dilution by impurity ions was found as a result of $2 - 3$ fold decrease in the FI losses. Improvement in the FI confinement at $B_t = 1.0$ resulted in longer characteristic time of T_i increase ($20 - 25$ ms instead of $10 - 15$ ms) and larger T_i increase (T_i increase at $B_t = 1.0$ is 50% larger than T_i increase at $B_t = 0.68$ T). With available power of 0.45 MW maximal obtained $T_i(0)$ was 0.36 keV. Thus, the clear positive influence of increased B_t on plasma heating by NBI was concluded.

EXC/P8-11

Fueling Control for Improving Plasma Performance in Heliotron J

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Control of fueling and recycling is one of the key issues to obtain high density and high performance plasma in magnetic confinement devices from two aspects; (1) profile control of core plasma and (2) reduction of neutrals in the peripheral region. This paper discusses the effects of the fueling control on plasma performance in Heliotron J, a flexible helical-axis heliotron device with an $L/M = 1/4$ helical coil. (The major and minor averaged radii are 1.2 m and 0.17 m, respectively. The averaged strength of the confinement field along the minor axis is < 1.5 T.) By using an supersonic molecular beam injection (SMBI) for fueling control, the stored energy reached ~ 4.5 kJ in a heating combination of ECH and Co-NBI, which is about 50% higher than the maximum one achieved so far under the same heating condition with conventional gas-puff fueling in Heliotron J. This study points out the SMBI method can avoid the confinement degradation in higher density region in the helical system. Deeper penetration and local fueling with a short pulse by SMBI can increase the core plasma density avoiding the degradation probably due to the edge cooling. In order to use more effectively this potential of SMBI, it is necessary not only to refine the SMBI system but also to control/reduce recycling.

EXC/P8-12

Comparison between Dominant NBI and Dominant ICRH Heated ELMy H-Mode Discharges in JET

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The extrapolations from present day machines to the ITER $Q = 10$ inductive standard scenario are predominantly based on ELMy H-modes heated by positive neutral beam (NB). The comparison with H-modes with predominant ICH heating can therefore help to confirm the prediction for ITER with a heating source that does not provide toroidal momentum input or fuelling and that can be centrally deposited at high density. Previous experiments comparing predominantly ICH and NB heated H-modes consistently reported no substantial difference in energy confinement enhancement factor H. Limited comparative pedestal analysis was available, and with contrasting results. In JET, the difficulties in coupling ICH with Type I ELMs resulted in a limited number of ICH heated H-modes in this regime. The scope of the JET experiment reported here is therefore to compare pedestal/ELMs, core confinement and transport of Type I ELMy H-modes with ICH and NBI in a relevant H-mode operational space in terms of plasma current, I_p (2.5 MA), q_{95} (3.6) and density ($60 - 70\%$ of n_G). 6 to 8.5 MW of ICH are reliably coupled with Type I ELMs using H minority heating at $B_t = 2.7$ T with 42 MHz dipole. The comparison of Type I ELMy H-modes heated only by NB and with different proportions of ICH (from 50:50 ICH:NB to 100% ICH) at different levels total power shows similar H factors. At constant power and density, core density and temperature profiles with ICH and with NB heating are similar, although the power is deposited mainly off axis with NB, and mainly on axis with ICH. In both cases $T_e \cong T_i$, although some increase in the temperature profile inside the sawtooth inversion radius is observed with predominant ICH. The toroidal rotation is ~ 10 times lower with predominant ICH than with NBI. The results are consistent with an effect of rotation on stiffness of the ion temperature profiles in the inner half of the plasma since high stiffness of temperature profiles in the ICH case keeps them close to threshold in spite of the high core heat flux, whilst the NBI core heat flux is much lower due to broad deposition. The pedestal pressure is similar with NB and with ICH for the same loss power and it increases with power in both cases. The pedestal widths of density, electron temperature and pressure are independent of additional heating mix.

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EXC/P8-13

Impurity Transport in TCV: Neoclassical and Turbulent Contributions

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Carbon impurity density profiles in TCV L-modes are studied in detail with respect to the role of sawtooth activity, of plasma current, of central ECH, of collisionality and of plasma shaping by comparing positive and negative triangularity. The carbon transport is compared to electron transport. It is shown that the carbon density profiles can have local normalized gradient much higher than the electron density, by a factor of two. However this is true only in the core of the plasma and outside the sawtooth inversion radius. Indeed, sawtooth activity has a strong influence, which can lead to an apparent dependence on plasma current. However dedicated experiments at different plasma currents I_p show that at radii outside the influence of the sawtooth activity, particle transport does not depend on I_p. Experiments with plasma currents between 105 kA and 345 kA with a inversion radius varying from 0.2 to 0.6 of the minor radius, have been analysed in details. The local density scale lengths are similar for electrons and carbon outside ($\rho \sim 0.6$). They are shown to be very different inside this value except within the area influenced by the sawtooth activity.

Adding ECRH in the centre flattens both profiles and it is used to study the effect of collisionality. This has also been performed in two types of discharges with different triangularity, positive and negative. It has been shown that the electron heat transport is reduced significantly in negative triangularity and across a large collisionality range. It will be shown that similar results are obtained for both electron and carbon particle transport

These results are analyzed with quasi-linear electrostatic gyrokinetic theory, which has been shown to explain relatively well L-mode electron particle transport in TCV. It is shown that both ITG and TEM modes play an important role. On the other hand the quantitative comparison for impurity transport is still challenging and the main issues regarding both neoclassical and turbulent transport will be discussed.

EXC/P8-14

Observation of Reduced Core Transport Triggered by ECRH Switch-off on the HL-2A Tokamak

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Anomalous transport is one of the important topics in fusion plasmas as it degrades the overall confinement. Many experiments have been performed on various tokamaks to clarify the formation of the transport barrier with reduced transport triggered by localized heating and fuelling, such as pellet injection, high Z impurity injection and electron cyclotron resonance heating (ECRH). The links between the improved confinement and the current profile, the q-profile and the magnetic shear have been studied in many literatures. A large central electron temperature increase is induced by off-axis ECRH in DIII-D, which was modeled in terms of a significant heat pinch and the suppression of the heat diffusivity. Observations in T-10 and TEXTOR have shown that the necessary condition for the reduced transport correlates with the appearance of the low value of $dq/d\rho$ in the vicinity of rational q-surfaces. However, it is still unclear about the links between the improved confinement, the ECRH power deposition and the core turbulence.

In this work, experiments with various ECRH depositions by changing the toroidal field have been performed on HL-2A tokamak. It is found that immediately after the far off-axis (near $q = 2$ surface) ECRH switch-off, the core electron temperature significantly increases for several tens of milliseconds before it starts to decrease, while the edge electron temperature rapidly decreases after switch off ECRH. The core temperature starts to decrease followed by the enhancement of the sawtooth activities. The increment of the core electron temperature appears inside the $q = 1$ surface. This corresponds to the formation of an internal transport barrier near $q = 1$ surface. The delayed decrease of the core electron temperature is observed with the ECRH deposition radius between $q = 1.5$ and $q = 2$ surface. This is similar to the observations in T-10 tokamak. The delayed decrease of the core electron temperature becomes small when the ECRH deposition radius is near or inside the $q = 1$ surface. Therefore, the transport barrier weakens as the ECRH power deposition moves from $q = 2$ to $q = 1$ surface. Measurements of the core density fluctuations near the transport barrier with an O-mode Doppler reflectometry show a clear reduction in the fluctuation level immediately after the ECRH switch-off.

EXC/P8-15

High T_e, Low Collisional Plasma Confinement Characteristics in LHD

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Since 2006, the installation of 77 GHz gyrotrons with each output power of over 1 MW has progressed in the Large Helical Device (LHD). These high power gyrotrons enabled us to achieve a higher electron temperature (T_e) than that previously obtained and also to survey properties of high T_e plasmas in wide configuration range due to the oscillation frequency selected as 77 GHz, which is different from those of the gyrotrons already installed in the LHD. High T_e plasmas of more than 15 keV were achieved due to the upgraded electron cyclotron resonance heating (ECRH) system. The value of T_e greatly exceeds $\sim$ 10 keV, which was obtained in previous experiments. Highly accurate T_e profiles were successfully obtained by the accumulation of the intensity of Thomson scattered light with the three YAG lasers all injected together. The configuration dependence of high T_e plasma performance was investigated under the magnetic field strength $B_0 \sim 2.7$ T and the optimum configuration for achieving high T_e plasma was observed. These high T_e plasmas accompanied an electron internal transport barrier (e-ITB). Clear threshold of density-normalized ECRH power for an e-ITB formation was found. The T_e profile drastically changed from a flat one to a peaked one when the e-ITB formation occurred. Once the e-ITB formed, the central T_e increased with the proportional dependence of the square root of the density-normalized ECRH power. One of the important advantages of ECRH is that perpendicularly injected ECRH can produce plasmas without driven toroidal current. The dynamics of the e-ITB formation and the relation between the foot point and the lower-order rational surfaces were investigated without the effect of inductive return current by using perpendicularly injected ECRH alone. We observed that the foot point of an e-ITB moved outward during ECRH injection. T_e at the core region increased with the foot point moving outward and the increase of the T_e gradient in the e-ITB region. The local flattening in the T_e profile gradually healed with the propagation of the e-ITB foot point. The foot point finally reached to a lower order rational surface of 0.5. We observed the T_e profile behavior in the electron density range up to 0.73×10^{19} m^{-3}.

EXC/P8-16

Edge-Core Interaction Revealed with Dynamic Transport Experiment in LHD

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Large scale coherent structures in both core and edge regions and their dynamics are clearly observed in a nonlocal transport phenomenon (NTP: a core T_e rise in quick response to edge cooling) on LHD. In this case, the low-density plasma, not having a distinguishing structure such as an internal transport barrier, is heated by 1 MW ECH and 2 MW NBI. Right after the edge cooling due to a TESPEL injection, a jump of dT_e/dr is found to take place in the region extending from $r/a \sim 0.6$ to at least $r/a \sim 0.7$). Although this dT_e/dr jump is surely affected by a jump in electron density due to the TESPEL injection, the increased dT_e/dr is sustained for a while unlike the usual case. Thus, a first order transition of electron heat transport, which is categorized by a discontinuity in dT_e/dr, appears over a wide region (at least 6 cm wide) in the periphery of the plasma. At about the same time, a second order transition of the electron heat transport, which is characterized by a discontinuity in the $d(dT_e/dr)/dt$, appears over a wide region (corresponds to a width of about 10 cm) in the plasma core. These indicate the existence of large scale coherent structures in both core and edge regions, which are of a scale larger than a typical micro-turbulent eddy size (a few mm in this case), and their interaction can cause the nonlocal T_e rise. In the plasmas with the NTP, a macro-scale turbulent structure has been observed in both the reflectometer and ECE signals. This macro-scale turbulent structure could support the interaction of the large scale coherent structures.

EXC/P8-17

Comparison of Pedestal Characteristics in JET & JT-60U Similarity Experiments under Variable Toroidal Field Ripple

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The pedestal characteristics of JET and JT-60U H-mode plasmas were compared in a series of similarity experiments. Despite of a very good match in the plasma geometrical parameters, essentially different number and shape of TF coils results in a different TF ripple amplitude and topology. In the similarity experiments performed at the lower I_p of 1.1 MA, no clear difference in the pedestal pressure was seen for the variation of TF ripple values for both devices. The toroidal rotation profiles were strongly affected by ripple. However, the changes in n_e, T_e and T_i profiles are unclear not only at the plasma edge but also in the plasma core. Generally, densities without gas puff are in range of $40 - 50\%$ of n_{GW} at ~ 1 MA in JT-60U. In JET, the achievable density without gas puff decreased with ripple at 2.6 MA while the density remains almost constant over the variation of ripple at 1.1 MA. This result implies the effect of TF ripple on density pumping might become stronger at higher I_p/B_t.

EXC/P8-18

Energy Confinement and Pellet Fuelling in MAST

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First applications of spherical tokamaks (ST) are foreseen as intense volume neutron sources such as a Component Test Facility (CTF-ST). The MAST H-mode confinement database shows that the extrapolation from MAST to CTF-ST is largest along the normalised collisionality and the second largest gap is towards the lower values of safety factor. The scaling of heat transport with these dimensionless parameters is investigated using dedicated scans in NBI heated H-mode plasmas where only one dimensionless parameter is varied while others are kept constant. A factor of four collisionality scan is

realised. The thermal energy confinement time shows a favourable scaling (better confinement towards lower collisionalities). Local heat transport analysis shows that the single fluid heat diffusivity is consistent with the global energy confinement scaling. The scaling of electron heat diffusivity shows a weaker collisionality dependence. It is noticed that along the collisionality scan, the neutron rate scales as the cube of toroidal magnetic field, in line with the scaling of beam slowing down time. A factor of 1.5 dimensionless safety factor scan is also realised. The scan indicates that the scaling of energy confinement time with safety factor is weaker than the cubic dependence in the $IPB98_{(y,2)}$ scaling. The gyrokinetic codes GS2 and GYRO are being used to analyse both dimensionless scans. Particle transport and fuelling is studied using high field side pellet deposition into NBI heated H-mode plasmas. Pellet-triggered Thomson scattering allows documentation of the density and temperature evolution during the pellet deposition and post pellet decay by up to 8 temporal points. High spatial resolution (~ 1 mm) visible bremsstrahlung imaging of the pellet trajectory reveals discrete structures (striations). The wavelength of the striations is in the range of $\sim 4 - 15$ mm with possible indication of correlation with bremsstrahlung emission amplitude. The pellet triggered Thomson scattering system also allows documentation of the pellet retention time.

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EXC/P8-19

Impact of Collisionality on Fluctuation Characteristics of Micro-Turbulence

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There are two prevailing approaches to confinement prediction for future devices such as ITER. The first, whose accuracy is rather limited, is to extrapolate global empirical scaling laws to these devices. The second approach, which is in principle more sound, is to use first principle models such as gyrokinetic simulations in integrated modelling. In this latter approach, validation of physics ingredients of these codes against experimental measurements is of critical importance. With this motivation in mind, dedicated dimensionless scan experiments have recently been performed on Tore Supra to investigate the impact of rhostar and nustar on transport and general turbulence characteristics. The present paper focuses on the impact of changing nustar on wavenumber spectrum and on phase velocity of density fluctuations measured using Doppler backscattering system. High resolution measurements exhibit perpendicular wavenumber spectra following a power law with a spectral index equals to -3 at low k and starting to decrease much faster for higher k. At basic level, the form of the k-spectrum appears to be robust and all spectra measured on Tore Supra are well represented by an exponential function or by a function derived from a simple drift wave turbulence model with disparate scale interactions. However, we show that the collisionality affects the shape of the spectra: parameters of both functions depend significantly on nustar pointing to the possibility of a transition in the turbulence regime. In addition to the wavenumber spectrum, Doppler backscattering system grants access to mean perpendicular velocity of density fluctuations which is the sum of the phase velocity of density fluctuations and of the mean $\mathbf{E}\times\mathbf{B}$ rotation. The dependence of mean perpendicular velocity of density fluctuations with the perpendicular wavenumber gives an indication about the dispersion relation since the mean $\mathbf{E}\times\mathbf{B}$velocity is independent of the wavenumber. It is found that in the high nustar discharges, the mean perpendicular velocity of density fluctuations decreases with increasing k while there is no clear dependence on k for the low nustar discharges. The existence of these two different forms may correspond to a transition between two modes with different dispersion relations as well as between a pure and a mixed mode.

EXC/P8-20

Particle Transport Investigation in HL-2A using ECRH and SMBI

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ECRH modulations have been carried out in HL-2A. The positive density perturbation due to the out-gassing propagates from the edge to the center. The negative density perturbation (pump-out) propagates from the center to the edge. This strong asymmetry in D and V for the negative and positive density perturbation looks similar to that the positive density perturbation is due to the particle propagation, the negative density perturbation is due to the turbulence spreading.

EXC/P8-21

Fluctuation Suppression during the ECH induced Potential Formation in the Tandem Mirror GAMMA 10

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The particle flux evaluated from the phase difference between the potential and density fluctuations measured by a gold neutral beam probe (GNBP) showed the radial outward flux and its correlation with the decrease in the stored energy. The suppression of the potential fluctuation was clearly observed during the radial potential profile change and positive electric fields driven by electron cyclotron heating (ECH). When flux is suppressed by the potential formation, the quick plasma stored energy increase is observed. These indicate that the observed fluctuations are the one of the source of the anomalous radial

transport and the radial profile of the potential is one of the keys to control plasma transport.

EXC/P8-22

Fuelling Efficiency and Penetration of Supersonic Molecular Beam Injection in HL-2A Tokamak Plasmas

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The penetration characteristics of supersonic molecular beam injection (SMBI) have been studied on the HL-2A tokamak. The signals from the tangential D_α array and CCD camera clearly show that the SMBI from low field side (LFS) consists of a slow component (SC) and a fast component (FC). The FC can penetrate more deeply than the SC, such as 8.5 cm inside the last closed flux surface (LCFS), while the SC penetration is around 4 cm. The penetration depth of the SC is weakly dependent on the line-averaged plasma density before injection and its backing pressure. The typical fuelling efficiency of LFS SMBI is 30 $\sim$ 60% for the limiter configuration. It is more the variation of the decay time of the post-SMBI electron density than the different injection depth that is responsible for the large scatter of the measured fuelling efficiencies. The fuelling efficiency and response of plasma to injection are compared among Gas puffing (GP), LFS and HFS SMBIs. The fuelling efficiency from high field side (HFS) is higher than that from LFS for the Ohmic heating discharges; moreover, with ECRH power of 1.3 MW, this trend becomes much more distinct indicating it is better for SMBI to fuel from HFS. And the LFS SMBI features relatively high fuelling efficiency and low perturbation to plasma.

EXC/P8-23

Perturbation Propagation in Laser Blow-off Impurity Injection in the TJ-II Stellarator and its Transport Results

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We have performed impurity injection experiments of low Z materials, deposited by two different sources, in TJ-II plasmas with the purpose of widening the scope of the transport studies to which can be applicable this technique and to compare these results with experiments performed with Fe injection.

The global confinement of Boron injected by the laser blow-off technique has been studied in both ECRH and NBI phases and similar confinement behavior has been observed as compared with Fe. Transport analysis is being carried out for selected shots and results will be presented.

The parallel transport has been studied by following the transient evolution of the perturbation, apart from the more standard diagnostics, with two filterscopes allocated 180°, which track the displacement of the injected material by monitoring a prominent line of the B^+. The propagation studied as a function of iota magnetic configuration, has been measured faster than expected.

Finally, the perturbation produced by the injection in other parameters is studied with spatial resolution in order to constrain the actual deposition radius of the injection, very critical for the transport analysis, and to follow the small controllable cooling of the plasma achieved with low Z materials to figure out if the electron transport can be inferred from these measurements.

EXD — Oral

(*Magnetic Confinement Experiments: Plasma - material interactions - divertors, limiters, SOL*)

EXD/2-2

Modification of Edge Profiles, Edge Transport, and ELM Stability with Lithium in NSTX

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The use of lithium in NSTX has enabled access to a high pedestal pressure regime, one in which the core stability limits at high normalized beta are observed with no sign of ELMs. Research in such ELM suppressed or small ELM regimes is of increasing importance for ITER, which requires that the projected fractional stored energy drop per ELM be maintained below 0.3%. Following application of lithium onto graphite plasma-facing components in NSTX, ELMs were eliminated gradually through growing periods of quiescence, with the resulting pressure pedestal widths increasing substantially. The modification of the pressure profile originated mainly from reduced recycling and edge fueling, which relaxed the edge density profile gradients inside the separatrix, effectively shifting the profile inward by up to $2-3$ cm. In contrast, the edge electron temperature profile was unaffected in the H-mode pedestal steep gradient region at constant plasma stored energy; however, the region of steep gradients extended radially inward by several cm following lithium coatings. Consequently, the pressure profile width increased substantially, with the peak post-lithium ELM-free discharge gradients comparable to the pre-lithium ELMy discharge gradients. Simulations of the measured edge profile changes with the SOLPS code indicated that both a reduction in recycling and a drop in the edge and SOL cross-field transport was required to match the post-lithium profiles.

Calculations with the PEST and ELITE codes have confirmed that the post-lithium discharge pressure profiles were farther from the stability boundary than the reference pre-lithium discharges, which were relatively close to the kink/peeling boundary. Indeed

low-n ($n = 1 - 5$) pre-cursors were observed prior to the ELM crashes in the reference discharges, consistent with the PEST and ELITE predictions. The resulting post-lithium discharges were ELM-free with a 50% increase in normalized energy confinement, up to the global $\beta_N \sim 5.5 - 6$ limit. While these ELM-free discharges ultimately suffer radiative collapse, pulsed 3D magnetic fields can be used to trigger ELMs on-demand to control density and purge impurities as needed.

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EXD/6-1

Analysis of Tungsten Melt Layer Motion and Splashing under Tokamak Conditions at TEXTOR

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Tungsten(W) is foreseen as the plasma-facing component (PFC) material for the ITER Divertor in the activated phase and proposed for DEMO. The main challenges of high-Z PFCs are the plasma radiation losses and the possibility of melting under uncontrolled conditions, posing additional large constraints for the power handling. Melting can lead to large W influxes into the plasma, reduce the lifetime of the PFCs and degrade the power handling capability due to subsequent surface irregularities. Melt layer dynamics and splashing are investigated mainly in plasma gun and electron beam experiments, with limited data from tokamaks. A dedicated R&D programe has been started under the IEA Implementing Agreement on Plasma Wall Interaction in TEXTOR. Sets of castellated W plates with different gap width and shaping were exposed in the TEXTOR PWI test facility to power fluxes of typically 30 MW/m^2. The main objectives were to analyze the formation of the melt layer, its motion and stability under the plasma impact and the magnetic field (2.25 T), in particular with respect to the possible bridging of the gaps and melt splashing. The melting of W led in all cases to a large material redistribution with the liquid W moving perpendicular to the magnetic field. The motion under TEXTOR conditions is driven mainly by a $\mathbf{j} \times \mathbf{B}$ force, with the current determined by the thermionic emission of the molten W. The force exceeds the plasma pressure force significantly. The motion of liquid W caused during one single melt event a restructuring of the surface with hills at the end of the castellation of up to 0.5 mm in height. However, no bridging of the gaps by molten W was observed. In a longer melt event (~ 2 s) with higher impact power

W droplet formation and ejection occurred. Melt droplets were ejected into the plasma and charged, returning them to the W target. Post mortem analysis near the molten area found W droplets with sizes between 4 μm and 100 μm and an estimated mass of 2 mg in total. Splashing led to a periodical accumulation of W in the plasma core leading to first minor then to a major disruption. The results of the melt layer behavior and material redistribution were used to benchmark the MEMOS-3D code.

EXD/6-2

Erosion and Confinement of Tungsten in ASDEX Upgrade

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In the fully tungsten clad ASDEX Upgrade, the erosion of tungsten, the penetration into the confined plasma and the transport of W in the edge transport barrier (ETB) of H-mode plasmas has been studied. The edge W dynamics in H-mode plasmas with type-I ELMs is to a large extent governed by the ELMs. The erosion at the plasma facing components strongly increases within an ELM, mainly due to sputtering by light impurities. The W sputtering within an ELM is predominantly located in the divertor, however, also the main chamber elements contribute $10-20\%$. The W source at the outboard limiters has the strongest influence on the W density in the confined plasma and the confined W density decreases with rising ELM frequency. A modelling approach with the 1D radial impurity transport code STRAHL was employed to calculate the erosion and edge transport of W eroded at the limiters.

One important modelling element is the prompt redeposition of W, i.e. the immediate return to the surface during the first gyration after ionisation. During an ELM, the increase of electron density and temperature in front of the surface causes a decrease of the ionisation length and prompt redeposition increases. Thus, the strong increase of the source is buffered. The second important element is the radial impurity transport in the ETB. Here, the density profile evolution of light impurities was measured with charge exchange recombination spectroscopy. It was found that between ELMs, all impurities are subject to a strong inward pinch leading to steep impurity density gradients in the ETB which are flattened during an ELM. The evaluated transport coefficients are in accordance with neo-classical theory and cause an increase of the peaking with rising impurity charge. It is strongest for W and leads to an approximately 3 times larger gradient in the ETB than for C.

The transport model was applied to ASDEX Upgrade discharges, where the ELM frequency was varied between 50 and 200 Hz by increasing the gas puff by a factor of 10. The W density decreased with rising ELM frequency and the model reproduced the measurement within 10%. It seems, that the model contains the dominant mechanisms to understand the edge transport of W. This is encouraging, because the underlying physics effects can in principle easily be calculated for ITER.

EXD/6-4

Particle Control and Transport Experiments in the DIII-D Tokamak with Graphite Walls

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Retention of hydrogenic isotopes in the first wall of future burning plasma experiments such as ITER can lead to an unacceptable in-vessel tritium inventory. To address this topic, we report new results from recent particle control and transport experiments in the DIII D tokamak. We find that dynamic particle balance calculations (particle sources and sinks calculated vs time) yield similar results to a shot-averaged "static" (calculated by pressure rise) particle balance. The dynamic particle balance measurements show very low wall retention in H-mode, compared with large retention in L-mode; ELMing H-mode discharges with either NBI or EC heating show similar results. Particle balance in discharges with resonant magnetic perturbation (RMP) ELM suppression show that the wall retention rate and inventory is dependent on pedestal density and divertor conditions. The dominant term in the particle balance equation during RMP discharges was the cyro-pump exhaust rate. ^{13}C injection experiments have shown that most of the carbon is deposited at the inner strike point of a SN divertor in L-mode, additional deposition in the private flux zone is present in H-mode. With an unbalanced DN plasma shape, there is more localized ^{13}C deposition near the injection point in the non-active divertor. Moderate heating of graphite DiMES samples to 200°C results in a factor of 10 less deposition compared to room temperature. A new Porous Plug Injector has been used to demonstrate that chemical erosion in the graphite divertor target is dramatically reduced in a cold, detached divertor. Preparations for a DIII-D demonstration of removal of re-deposited carbon with an oxygen bake have progressed and final reviews are in progress. An oxygen bake of 2 Torr for 2 hours at 350°C is predicted to remove deposited ^{13}C layers in DIII-D. In ITER, removal of the carbon would also remove the co-deposited tritium. A new technique involving processing previously exposed and analyzed DIII-D tiles "known to be rich in ^{13}C" would be processed in DIII-D. These could be installed during a clean vent. This tests ^{13}C removal in "remote" regions of the torus, and provides a better test of restart of plasma operations after the oxygen bake.

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EXD/6-5Ra

Edge Impurity Transport Study in Stochastic Layer of LHD and Scrape-off Layer of HL-2A

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Edge impurity transport is studied in LHD and HL-2A tokamak as a comparative analysis based on carbon emission profile measurement and edge transport code EMC3-EIRENE. The parallel (//) motion of impurity is mainly determined by the balance between the friction caused by background plasma flow and the thermal force, while the perpendicular motion is assumed to be purely diffusive. The code shows that the friction force dominates over the thermal force at high density (n) range and the impurity is pushed back towards downstream. In such case, impurity density profile reflects the poloidally localized divertor leg structure.

The effect is visible in CIV intensity profiles obtained by the 3D code with an intensity peak appearing around the divertor leg at high n. The CIV profiles measured with EUV spectroscopy clearly confirms this impurity movement. The screening effect is measured with ratio of CV, as proxy for impurity ions at deeper radial position, to CIII+CIV, as proxy for source. In LHD the ratio of emission monotonically decrease with increasing SOL collisionality (SC) of up to 70. The code can reasonably reproduce the tendency. In HL-2A the screening effect also appears in the ratio, but it saturates at lower SC of ~ 10. This is due to the narrow operation range of SC < 20 limited by detachment. The downstream of the HL-2A SOL changes into extremely collisional regime, which leads to the detachment, while the thermal force is still dominant at the upstream. This is because of the strong coupling between downstream n & temperature (T) and upstream n in tokamak SOL, known as high recycling regime. The higher collisionality at the upstream in friction dominant LHD edge plasma is allowed by the modest dependence of the downstream n & T on upstream n, due to the momentum loss of //-plasma flow induced by the strong perpendicular interaction of stochastic flux tubes. The suppressed downstream n (lower than upstream n) provides deeper penetration of recycling neutrals to shift start point of flow acceleration upstream. The remnant island structure is found to reduce //-T gradient due to the very small ratio B_r/B_t, which replaces partly //-energy flux with perpendicular one. The perpendicular transport caused by the stochastic magnetic field is found to be a possible reason for the differences in the edge impurity transport between the two devices.

EXD/6-6Rb

Power Load Characterization for Type-I ELMy H-Modes in JET

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Following the installation of a fast high resolution infrared (IR) system viewing the JET divertor and using the existing wide-angle IR-camera a large number of experiments have been performed optimized for power exhaust studies with focus on ELM and inter-ELM profiles. Type-I ELMy H-mode deuterium plasmas with currents from $1 - 3.8$ MA, $q_{95} = 3.3 - 5.4$ with low and high triangularity were investigated. The IR based absolute estimate of the surface temperature of the CFC-tiles was validated by comparing with tile embedded divertor thermocouple (DVTC) and pyrometer temperature measurements. The discharge integrated energy balance from IR with respect to DVTC is found to lie within a range of 80-120%. Comparison to divertor Langmuir probe measurements are used to assess the influence of the divertor target surface thermal properties on the inferred heat fluxes by IR for short events such as ELMs. The energy balance between the ELM loss, divertor and outer limiter ELM thermal load, injected and radiated power during a time interval of 4 ms is studied.

The wetted area of the inter-ELM profile varies from 0.6 m^2 to 0.35 m^2 within the data base, corresponding to mid plane e-folding decay lengths of 6mm and 3.5mm. The ELM target wetted area shows an increasing broadening with ELM size. The dynamics of ELM imprints at the target caused by ELM filaments was studied for individual ELMs. It is found that in the rise phase of the ELM target power the number of these striations increases from $3 - 5$ to numbers around $10 - 20$. The striation number seems to be independent of the input power. The increase of the ELM target heat load in an early stage up to 40% ELM rise time is associated with an increase of the number of single filaments. For later times the number of observed striations remains constant, but the striation magnitude changes. The impact of ELM filaments on the first wall has been studied by varying the distance of the separatrix to the outer wall limiter on a shot-to-shot basis. The ELM mid plane far-SOL radial power decay length is found to be

$\sim$ 20 mm. It is observed that larger ELMs deposit a larger fraction of their energy onto the limiters than smaller ELMs. This fraction of ELM energy deposited on the outer wall limiters was found to increase roughly proportional to the square root of the normalized ELM loss energy.

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EXD/8-2

Magnetic Perturbation Experiments on MAST using Internal Coils

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In order to assess the ability of Resonant Magnetic Pertubations (RMPs) to mitigate type I ELM power loads, experiments have been performed on MAST using internal ($n = 3$) resonant magnetic perturbation coils. Calculations of the magnetic perturbation to the plasma due to the coils (either external or internal) have been performed using the ERGOS code (vacuum magnetic modelling). The Chirikov parameter (σ_{ch}), which is a measure of the island overlap, is used to define the stochastic layer as the region for which σ_{ch} is greater than 1. Although it is not clear if vacuum modelling or indeed stochasticity is relevant, this parameter does allow a way to compare different configurations.

Experiments on L-mode plasmas, with a plasma current $I_p = 400$ kA, show a clear influence of the coils. A strong density pump-out is observed (with a drop in ne of up to 30%), together with an increase in the turbulent fluctuation level and change in the radial electric field. This effect is seen only when the coils are in the configuration (even versus odd parity) predicted to be on resonance with the plasma. In H-mode plasmas some effect was demonstrated for plasmas just above the L-H transition threshold: the application of the coils seems equivalent to a small decrease in input power. For example, the application of the RMPs can trigger type III ELMs in an ELM free discharge or trigger more frequent ELMs in a type III discharge. Initial experiments showed that the application of the RMPs to discharges with type I ELMs produces little effect on the ELM behaviour or pedestal characteristics. This is despite the fact that vacuum modelling shows that the Chirikov parameter is greater than 1 for a radial extent wider than that correlated with ELM suppression in DIII-D. However, more recent experiments have shown that it is possible to increase the ELM frequency and decrease the ELM energy loss by carefully adjusting the q_{95} of the plasma.

These observations will be compared with the results from non-linear MHD modelling, which show that a strong rotational screening takes place in most of the pedestal.

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EXD/P3 — Poster Session

EXD/P3-01

Divertor Profile Modification by the Effect of 3D Field Perturbation in NSTX

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Externally applied 3D fields have been used to suppress or trigger ELMs in tokamaks and are considered to be used in ITER. It is found that the non-axisymmetric magnetic perturbations modify the heat and particle flux profiles at the divertor surface in NSTX H-mode plasmas. They produce multiple strike points radially separated, formed by the so-called "lobe" structure. A modest level of strike point (SP) splitting is observed even before the application of 3D fields for some NSTX discharges, i.e. the "intrinsic" SP splitting, for which several possible sources such as the intrinsic error fields, MHD modes, and the toroidal field ripple can be thought as the origin. Divertor profiles broaden when the plasma enters the intrinsic SP splitting phase, with the peak values at the separatrix largely unchanged. The intrinsic SP splitting is augmented by the externally applied 3D fields. This "augmented" SP splitting by the applied 3D fields was simulated by a field line tracing code and was compared with the measurement for $n = 1$ and $n = 3$ cases. It is found that the location and spacing of the simulated split strike points agree well with the observation. The inclusion of plasma response inside the separatrix in the field line tracing did not substantially alter the predicted footprints at the divertor targets. Time response of the formation of the augmented SP splitting is as fast as $3 - 4$ ms, which is consistent with the field line penetration time through the vacuum vessel. The pedestal electron temperature and density profiles show quick (< 15 ms) reduction by the applied 3D fields but the electron collisionality is not found to change significantly. The SP splitting observed in the heat flux profile in NSTX occurred in the pedestal e-collisionality of $1 - 2$, which is consistent with the DIII-D result, where it is only observed for e-collisionality > 0.5. Understanding the effect of 3D fields on the divertor profiles and the underlying physical mechanism of the profile modification is crucial for the development of effective control tools for the divertor heat and particle handling. This is particularly important for the future spherical and conventional tokamaks because of the extremely high heat load onto the divertor plates expected for those machines.

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EXD/P3-02

Experimental and Simulation Studies of Dust Transport in JT-60U Tokamak

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Understanding of the dust transport as well as determination of where it is ejected and finally deposited are of a great importance to predict the plasma performance and the tritium retention in a fusion reactor. Laser scattering measurement showed that huge number of carbon dust were observed in the main chamber of JT-60U at disruptions and in the following discharges. Radial distribution of the total dust number during and just after the plasma termination of the disruption was relatively uniform, suggesting that they were redistributed over the wide surface area. Many dusts were ejected also in the following discharge, where both the size and number were peaked in the far-SOL near the plasma facing component and they decreased near the separatrix. This result suggests that sublimation of dust is dominant even in the SOL. Lifetime until the sublimation and the distance from the separatrix were calculated for various size of dust, using DUSTT code, to be 2−200 ms and 22−4 cm with increasing the radius of 0.1−100 μm, respectively. Experimental and simulation results showed very few probability of the penetration into the confined plasma as an impurity source.

EXD/P3-03

Morphology Classification and Video Tracking of Dust Particles in ASDEX Upgrade

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Due to the relevance of dust particles in respect to safety and plasma performance, the morphological properties of dust particles as well as their generation, mobilization, transport and deposition pattern have to be investigated.

The main investigations at ASDEX Upgrade (AUG) related to dust are in-situ detection of plasma wall interaction areas and migration of strongly radiating particles during plasma operation using infrared, standard and fast cameras and post-mortem analysis of collected particles using optical light microscopy and automated analytical scanning electron microscopy (SEM with energy dispersive X-ray spectrometry (EDX)) assisted by cross-sectioning with a focused ion beam.

The automated EDX analysis of particles collected since the transition of AUG to a full tungsten plasma-facing wall allows to obtain statistically data containing thousands of particles down to particles size of $\sim$ 100 nm and to classify the particles. Even classes of particles with low abundance but distinct configuration (e.g. spherical particles from diagnostic mirrors) can be separated from morphologically comparable particle fractions, e.g. tungsten droplets produced by arcing. Furthermore, particles below 5 μm are dominantly spherically and show W as their main constituent, while above 5 μm complex agglomerates were found dominated by C and B matrix with embedded W spheres. Sampling method dependent collection efficiencies were considered as an important influencing factor for data interpretation.

The in-situ registered event trajectories are automatically classified into stationary hot spots, single-frame events (e.g. by neutrons) and real dust particle fly-bys with velocities in the range of $10-100$ m/s. A considerable decrease of detected events with progressing plasma operation was observed. After $\sim$ 1800 s operation, even deliberately induced disruptions do not induce more dust events. This indicates an absence of mobilizable dust at that time of operational period.

In addition to the above described investigations, the over-all strategy of investigations at AUG related to dust will be presented briefly and the rule of arcs on dust production will be discussed.

EXD/P3-04

Fuel Retention in Discharges with Impurity Seeding after Strong Be Evaporation in JET

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Preparatory experiments for the ILW were carried out to simulate the massive Be first wall by a thin Be layer, induced by evaporation of 2.0g Be, and to study its impact on fuel retention and divertor radiation with reduced C content and N seeding. The evolution of wall conditions and the increase of C radiation with constant N radiation have been studied in a series of ELMy H-mode plasmas (2.7 T, 2.5 MA, $P = 16$ MW). A reference discharge was executed prior to the evaporation to document the initial conditions. Residual gas analysis shows a reduction of CxDy ($\times$15) and O ($\times$6) in the partial pressure. The Be flux (BeII 436.0 nm) was increased by a factor 20 whereas the C flux (CII 426.7 nm) decreased by $\sim$ 50% in the limiter phase of the first discharge. Erosion of the Be layer and partial coverage with C takes place; the Be flux drops to a factor 5, the C flux increases moderately up to 60% of the initial value in 4 discharges.

To make use of the protective Be layer, only the first 4 discharges were used for a gas balance analysis of the D and N inventory. The major fraction of the recovered gas is D which leads to a retention rate of 1.94×10^{21} D/s comparable to retention rates with C walls. However, thisretention rate is overestimated, because Be evaporation provides a non-saturated surface with respect to D. Secondly, C was only moderately reduced and co-deposition with Be and C takes place. These transient effects observed here, wall loading and re-appearance of C, will not take place with the ILW. Co-deposition with Be will remain as mechanism for retention, though we assume that Be transport in remote areas will be reduced due to absence of chemical sputtering. About 3.0% of the gas recovered is N; equivalent to 30% of injected N. However, NII spectroscopy indicated a legacy of 25% which cannot account for missing N.

The reduction of C shall have an impact on the divertor radiation. However, the radiation in phases prior to N injection is only moderate decreased. N radiates close to the x-point whereas C radiates also along the separatrix. N adds to the radiation of remaining C and D and the N content increases due to the legacy effect. But C radiation starts to increase in time, and both contributors causing a continuous increase of the radiation fraction from 50% to 70%. The pattern indicates that N dominates the radiation increase in the first discharges.

EXD/P3-05

Spatial Structures of Plasma Filaments in the Scrape-off Layer in HL-2A

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Spatial structures of plasma filament in toroidal magnetized plasma are investigated with the novel combination of a poloidal Langmuir 10-probe array and a radial Langmuir 8-probe array, which are separated toroidally by 210 cm. The parallel wave number changes its sign and the coherency reaches maximum at the ideal intersection point of a magnetic field line passing by. Supposing the local safe factor is appreciatively calculated, we observed the filament distribution along the magnetic line by changing the magnetic pitch angle. Two inclined filaments have been firstly observed with the novel Langmuir probe array combination. One gradually disappears and another gradually grows and moves into our observation region, indicating the filament birth zone is inside the separatrix. The lifetime and propagation velocity of filaments are estimated to be 25 μs and $0.5-2.0$ km/s, respectively. Filament driven by interchange mode also had been studied.

EXD/P3-06

Lithisation Effects on Density Control and Plasma Performance in RFX-Mod Experiment

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Plasma-wall interaction is one of the most important issues that present magnetic confinement devices have to face. In the RFX-mod experiment density profile control recently became even more crucial point due to the observation that single-helical-axis states (SHAx), and their consequent improved confinement regime at high plasma current ($I_p > 1.2$ MA), spontaneously disappear when operating at medium plasma density ($n/n_G \geq 0.2$). It is believed that a good density profile control will allow to operate at higher density improving, at the same time, the energy confinement. Following tokamak experience, different techniques have been tested to improve density control: He glow discharges cleaning, high plasma current He discharges, wall boronisation and baking. All such techniques were able to improve the situation but none allowed a complete and stationary density control. At high plasma current their effect lasts only few discharges, characterized by a lower than usual Hydrogen influx but in any case high enough to reach at low density equilibrium with a recycling factor equal to one; furthermore, the wall often responded in an uncontrollable way providing sometimes very high influx during the discharges. In order to improve this situation, based on good results obtained on Tokamak and Stellarator experiments, we tested the effect of wall conditioning by Lithium.

As a first lithisation method to deposit on the wall a controllable amount of Lithium we used a room temperature pellet injector (max diameter of 1.5 mm and max length of 6). Lithisation was applied both directly to the graphite tiles and over a fresh boronisation. The technique of depositing Lithium by pellet injection is found effective in maintaining Hydrogen wall recycling very low. A good effect is observed also on edge density providing more peaked density profiles. Experiments of wall conditioning by lithium will be performed also by a Liquid Lithium limiter with capillary porous system on loan from FTU experiment of ENEA laboratories in Frascati. The effectiveness of the methods was firstly studied in terms of effect on the wall analyzing graphite samples inserted during conditioning operations in the same position of the graphite tiles. The ability in density control, evaluating the amount of hydrogen that the wall was able to adsorb, and the effect on plasma performance has been analysed.

EXD/P3-07

ELM Simulation Experiments on Pilot-PSI using Simultaneous High Flux Plasma and Transient Heat/Particle Source

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Edge Localized Modes (ELMs) in ITER will lead to material erosion, cracking, melting and vaporization. During an ELM, the divertor surfaces are submitted to both the steady state detached divertor plasma and the intense heat and particle fluxes during ELMs. Such a situation will lead to synergistic effects which might strongly affect the material damage threshold [1]. In this contribution we describe a new experimental setup for ELM simulation experiments with relevant steady-state plasma conditions and transient heat/particle source.

The Pilot-PSI linear device produces plasma parameters relevant to the study of plasma-surface interactions in the ITER divertor [2]. In parallel to the DC power supply, the plasma source is also connected to a capacitor bank (5 kV, 8.4 mF, 100 kJ) which is discharged in the plasma source to create a transient increase of the discharge current [3]. The pulse duration is about 1 ms. This allows the superimposition of a transient heat and particle pulse to the steady-state plasma. The plasma parameters are measured by Thomson scattering and the sample temperature during the pulse is monitored by a fast infrared camera. The plasma source can be operated in pulsed mode in a variety of gases (Ar, H, He, N) as well as with mixed gases.

So far, discharge currents of up to 11.6 kA have been achieved, which represents a peak power in the plasma source of about 5 MW. The highest plasma density of 1.5×10^{22} m^{-3} and temperature of 4.5 eV leads to a surface peak heat flux of 0.9 GW·m^{-2}. The temperature rise time is in the range $300 - 500\,\mu$s, comparable to that observed during Type-I ELMs in JET. With higher gas flow, it is anticipated that peak heat fluxes in the range $1 - 3$ GW·m^{-2} will be reached.

Exposure of tungsten targets to hydrogen plasma with transient revealed an abrupt increase of the plasma density after the pulse at the location of the TS system (17 mm from the target). The influence of such heat pulses on the deuterium retention in tungsten will be described. In addition, the pulsed plasma source has been used to assess the effect of transient heat/particle pulses on the growth of helium-induced nanostructure and characterize the generated dust.

[1] S. Kajita, et al., Appl. Phys. Lett., 91 (2007) 261501

[2] G. Van Rooij, et al., Appl. Phys. Lett., 90 (2007) 121501

[3] G. De Temmerman, et al., Appl. Phys. Lett, submitted

EXD/P3-08

Multi-Component Plasma Interactions with Elemental and Mixed-Material Surfaces

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While significant advances in the understanding of PMI have occurred in recent years, much of this understanding concerns the interaction of a single-component plasma with an elemental material surface. This should be contrasted to ITER where material migration from the differing plasma-facing surfaces will lead to the formation of mixed-material surfaces interacting with the incident plasma. In addition, a burning plasma in ITER will generate helium ash that will be included in the plasma reaching the surfaces surrounding the plasma. In present and proposed confinement devices, radiating species (such as argon, neon or nitrogen) are intentionally added to the divertor plasma and these radiating species will affect the PMI. This paper presents results showing that plasma-surface interactions (both erosion and retention properties) cannot easily be predicted by the superposition of results from pure, single-component plasma interacting with elemental surfaces. Small amounts of condensable impurities in the incident plasma can have dramatic consequences to the erosion behavior of the surface (for example, beryllium carbide surface layer formation), whereas small amount of non-condensable impurities (such as helium) have been observed to dramatically alter the retention of fuel species within the plasma-exposed material. The simulation plasma used to investigate plasma-material interactions must be configured to conform as closely as possible to the conditions expected in confinement devices, both with respect to composition of materials in a surface, as well as constituents of the impinging plasma, to understand the evolution of surfaces and the feedback of the resultant surfaces to the incident plasma.

EXD/P3-09

Effect of Magnetic Island on Three-Dimensional Structure of Edge Radiation and its Consequences for Detachment in LHD

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Detached plasmas represent an important operational regime for a fusion reactor whereby the heat load to the divertor can be reduced through enhanced radiation to ensure sustainable steady-state discharges. Especially in the Large Helical Device (LHD), with a nonaxisymmetric magnetic field configuration, it is important to understand the three-dimensional (3D) structure of the radiation region. Normally, in toroidal devices there exists a density threshold above which detachment occurs, but in LHD the plasma commonly experiences radiative collapse before this threshold is reached. Recent work on LHD has shown the addition of an $n/m = 1/1$ magnetic island (MI) enhances the detachment process by lowering its density threshold [1]. In this paper the effects of the MI on the 3D radiation structure in attached and detached plasmas as predictied by EMC3-EIRENE [2] are clearly seen in the imaging bolometer (IRVB) [3] data, experimentally confirming the role that the MI plays in the detachment process. Three plasma cases are shown: attached with and without MI at low density and at higher density with MI and detachment. With the addition of the MI the carbon radiation profile from the code in a poloidal cross-section becomes more localized near the helical divertor (HD) x-points (X). This is reflected in the focussing of the radiation patterns corresponding to the HDX in both the IRVB and code data in images corresponding to the IRVB field of view (FOV). Detachment results in a more asymmetric radiation profile in the poloidal cross-section code data with localized peaks near the HDX and MIX. The radiation from the MIX is reflected in strong radiation from the corresponding location in the IRVB FOV from both code and IRVB data. However the relative increase in the radiation from the MIX is greater in the code data than in the IRVB data for reasons which are so far unknown. Also similar discharges with and without the MI show detachment with the MI, albeit at a lower density thant the discharge without the MI. This work confirms the previous conclusion that the MI enhances the localization of the radiation and is conducive to achieving and sustaining the detachment [1].

[1] M. Kobayashi et al., accepted for publication in Phys. Plasmas.

[2] Y. Feng et al., Contr. Plasma Phys. 44 57 (2004).

[3] B.J. Peterson et al., submitted to Plasma Fusion Res.

EXD/P3-10

Deuterium Retention Mechanism in Tungsten-Coatings exposed to JT-60U Divertor Plasmas

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In ITER, full tungsten (W) divertor and W first wall, possibly W-coating, has been proposed to reduce in-vessel tritium (T) retention. The W-coating is simultaneously irradiated with hydrogen isotope and carbon (C) particles which remains in the vacuum vessel due to erosion of carbon fiber composite (CFC) in the initial operation phase. Therefore, in order to predict the T retention in W, it is essential to investigate the effects of C on T retention. In the present work, deuterium (D) retention mechanism in the W-coating exposed to JT-60U divertor plasmas was investigated. The D retention in the W-coated tiles within ~ 1 mm thickness was poloidaly uniform at $\sim 10^{22}$ D/m^2 at the incident ion fluence of $10^{25} - 10^{26}$ m^{-2}, the typical incident ion energy of ~ 250 eV and the maximum surface temperature of ~ 700 K. This D retention was higher than that observed in basic experiments under similar experimental conditions described above for JT-60U by more than one order of magnitude. This high D retention in the W-coated tiles exposed to JT-60U divertor plasmas was caused by D trapping with C, which was implanted and diffused from the surface of the W-coating during the plasma discharge. The D/C ratio in the W-coating was $0.04 - 0.08$ which was $1/4 - 1/2$ of T/C ratio in C deposition layer. Therefore, simultaneous use of C based armor materials and W-coating might increase T retention in W-coating in future DT fusion devices due to the implantation of C.

EXD/P3-11

Effect of Ion Mass and Charge on Divertor Heat Load Profiles on JET

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The physical mechanism determining the heat loads on divertor tiles remains elusive. One way of distinguishing between the various candidate mechanisms is to compare otherwise similar plasma discharges with different main ion mass and charge. In practice, such a comparison can be achieved for hydrogen (H; $A = 1$, $Z = 1$), deuterium (D; $A = 2$, $Z = 1$) and helium (He; $A = 4$, $Z = 2$) plasmas. Recently, such a comparison was attempted at JET based on a series of dedicated experiments in H, D and He plasmas, with identical magnetic equilibria. In each case, several engineering parameters were varied: the toroidal magnetic field (1 to 3 T), the plasma current (1 to 3 MA), and hence the edge safety factor (3 to 5), the neutral beam heating power (2 to 18 MW), the fuelling rate and hence the line average electron density and the corresponding Greenwald fraction (0.5 to 1). The key observation of interest for tokamak power exhaust are heat load profiles on the outer divertor target. These profiles were measured using an infra-red camera with high spatial (< 2 mm) and temporal ($\sim 80\,\mu$s) resolution, allowing separate measurement of inter-ELM, ELM, and time-averaged heat load profiles. The results were expressed in terms of empirical scalings of peak heat load vs both engineering and physics based parameters, and the obtained scalings were compared with those previously reported and as well as with predictions of a wide range of theoretical models. It was found that in typical, low-to-medium power ELMy H-mode discharges, inter-ELM and time averaged profiles were moderately broader ($\sim 50\%$) in H and He, compared to D, plasmas, with an associated reduction in the peak heat load. In contrast, the effect on ELM profiles was more benign, with comparable power width in D, H and He plasmas, with a much longer power arrival time scale in He compared, to either D or H, as expected based on smaller sound speed and higher collisionality. The above results carry both positive and negative implications for ITER: the former, in that the physics of inter-ELM edge-SOL transport could be studied in both H and He plasmas and than extrapolated to D-T plasmas, and the latter, in that the study of ELM exhaust physics may require access to ELMy H-mode in H (as opposed to He) plasmas, which at present appears challenging.

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EXD/P3-12

A New Explore: High Frequency Glow Discharge Cleaning in the Presence of Toroidal Field on EAST

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Wall conditioning of fusion devices involves removal of hydrogen isotopes and impurities from interior device surfaces to permit reliable plasma operation. Techniques used in present devices include baking, metal film gettering, deposition of thin films of low-Z material, pulse discharge cleaning, Glow Discharge Cleaning, Radio Frequency Discharge Cleaning, and in situ limiter and divertor pumping. Although wall conditioning techniques have become increasingly sophisticated, a reactor scale facility will involve significant new challenges, including the development of techniques applicable in the presence of a magnetic field and of methods for efficient removal of tritium incorporated into co-deposited layers on plasma facing components and their support structures. Recently, A NEW EXPLORE, High Frequency Glow Discharge Cleaning (HF-GDC), is innovated and demonstrated in the both of EAST and HT-7 Superconducting Tokamaks in ASIPP. Preliminary results show: uniform glow in toroidal direction with B_t, wide range of pressure with excellent stability and no arcing. The removal rates are similar with RF-DC and GDC. This method is not only useful for wall-conditioning, but also very helpful for pre-ionization of plasma startup. Primarily results will be shown in this presentation. The very uniform high-frequency glow was observed around toroidal direction in the central region from two of the CCD in the opposite direction, even though only one electrode. The region of HF-GD plasma is strongly relative to geometry and position of electrode. More wide rang of pressure about from several Pascal to 10^{-4} Pa was obtained, work gas are helium and deuterium. The parameters of HF-GDC power supply are $U = 2.0$ kV, $f \sim 25 - 100$ kHz, $I \sim 10.0$ A. Plasma performance for High frequency glow discharges was investigated by reciprocator Langmir probe w/o B-field. Furthermore, mechanism of high frequency glow discharge relative to E-field, vertical and horizontal field will be reported. Removal efficiency of impurities and hydrogen isotopic comparing with conventional GDC and RF-DC, design of special electrode for HF-GDC, optimization for parameters of power supply, e.g. $U/f/I$, will be discussed.

EXD/P3-13

Dependences of the Divertor and Midplane Heat Flux Widths in NSTX

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Previously it was found [1] on the National Spherical Torus Experiment (NSTX) that the lower divertor heat flux widths, λ_q^{mid} decreased with increasing I_p, and was relatively independent of P_{NBI} at high P_{NBI}. However, all of the previous results were limited to lower divertor triangularity $\delta \sim 0.5$. Here we have extended the studies to higher triangularity of $\delta \sim 0.7$, confirming and extending the previous trends up to 1.2 MA. At higher triangularity, the same relative trend is observed with a $\sim 50\%$ contraction in λ_q^{mid} when the plasma current is increased from 1 to 1.2 MA. For high triangularity discharges, a trend of increasing heat flux profile is shown when δ_r^{sep} is decreasing between -5 to -15 mm. While for low triangularity discharges λ_q^{mid} is observed to be relatively constant at ~ 10 mm over a wide range of δ_r^{sep}. This results in increased heat flux as δ_r^{sep} increases at high triangularity. λ_q^{div} shows as strong increase as the magnetic flux expansion, f_{exp} increased over a range from 10 to 40. This leads to a strong reduction of the peak heat flux, where the peak heat flux is reduced from 8 to 2 MW/m^2. Finally, NSTX utilizes unique lithium conditioning to condition its graphite PFCs [2]. This results in ELM-free discharges [3] leading to a further contraction of the heat flux profile. While uncertainty in the surface emissivity for lithium coated graphite plasma facing surfaces limits a direct comparison of the magnitude of the heat flux profiles, the relative change in radial profile shape from pure graphite to a lithium coated surface is still applicable. The reduced edge collisionality due to the lithium coating appears to lead to an increase in transport of thermal energy onto the divertor. The experimental details will be presented as well as the impact on the performance of future spherical tokamaks.

[1] R. Maingi, C.E. Bush, et al., J. Nucl. Mater. 363-365 (2007) 196-200

[2] M.G. Bell, H.W. Kugel, et al., Plasma Phys. Control Fusion **51**, Vol 12 (2009) 124054

[3] R. Maingi, T. Osborne, et al., Phys. Rev. Lett. **103**, Vol 7 (2009) 075001

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EXD/P3-14

Recent Advances in Long Pulse Divertor Operations on EAST

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EAST is a fully superconducting tokamak with a flexible poloidal field control system to accommodate both single null (SN) and double null (DN) divertor configurations, thus providing a unique platform to address power and particle exhaust issues for next-step fusion devices such as ITER and beyond. Significant progress has been made in EAST on both physics and engineering fronts toward steady state operations since the last IAEA Fusion Energy Conference. Long pulse diverted discharges well over 60 seconds have now been achieved in EAST by lower hybrid current drive with advanced wall conditioning, active divertor pumping using the newly installed internal toroidal cryopump, and real-time feedback control of plasma configurations using the iso-flux control technique developed by joint efforts from the DIII-D and EAST plasma control groups. The first systematic assessment of both SN and DN divertor performance has recently been made in EAST under Ohmic and L-mode operating conditions. Particle and heat fluxes at both inner and outer divertor target plates are lower for DN, as expected, but with stronger asymmetry favoring outer divertors. In addition, DN operation leads to an up-down asymmetry with greater power and particle fluxes to the divertors with their ∇B drift towards the X-point. Reversing toroidal field direction exhibits a strong influence on the observed divertor asymmetries. Contributions from various edge drifts are assessed using the SOLPS code with drifts and currents. To actively control power and particle fluxes at divertor target plates, localized divertor puffing of Ar has been carried out in EAST, leading to over 50% reduction of the peak heat flux at the outer divertor target plate without significantly affecting the core impurity content. Methane injection at various divertor locations (inner divertor, outer divertor and private region) has also been explored to quantitatively assess divertor screening for intrinsic carbon impurities. Further efforts will be dedicated to the active feedback control of gas puffing rates to maintain the divertor plasma in partial detachment conditions, which is essential for achieving higher power, long pulse operations during the next EAST experimental campaign starting in March, 2010. This presentation will highlight these advances. Comparisons will also be made with shorter pulse diverted tokamaks.

EXD/P3-15

Investigation of the Toroidal and Poloidal Dependences of First Wall Conditions in the Large Helical Device

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The non-uniform wall conditions such as the hydrogen retention and the erosion/ deposition were investigated in the Large Helical Device (LHD) by using toroidally and poloidally distributed material probes. The numbers of probes located at the first walls along the toroidal and the poloidal directions were ten and five, respectively. They were installed every experimental campaign from 2003 to 2009, and the evolutions of the wall conditions were clearly obtained. The non-uniform wall conditions and the evolutions significantly depended on the operational procedures and in-vessel devices positions. In the recent six campaigns, the amount of retained hydrogen amount was measured in its toroidal profile, and it is seen that the amount of retained hydrogen was large at the wall in the vicinity of anodes used for the hydrogen glow discharge. The amount of retained hydrogen also drastically decreased as increase of the campaign number. This reason is owing to the reduction of time period for the hydrogen glow discharge. These results shows that the hydrogen glow discharge has an important role for the hydrogen retention.The toroidal profile of the wall erosion was obtained. The eroded depth was observed to be large also at the walls close to the anodes. The inert gas (He, Ne) glow discharges were conducted in addition to the hydrogen glow discharge. These inert gas discharges significantly contributed to the erosion. In addition to the hydrogen retention and the erosion, the toroidal dependence of impurity deposition was investigated. Major species of deposited impurities were B and C, and the amounts of retained hydrogen in the walls with thick depositions of B and C were large. The deposition of B or C also changed the fuel hydrogen retention. Furthermore, the toroidal dependence of inert gas (He, Ne) retention will be shown in the presentation.In the present study, the positional dependence of the wall conditions was newly and systematically investigated. These results significantly have contributed to make an experimental plan for the next campaign.

EXD/P3-16

Dust Particles in Controlled Fusion Devices: Generation Mechanism and Analysis

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Generation and in-vessel accumulation of carbon and metal dust are perceived to be serious safety and economy issues for a steady-state operation of a fusion reactor, e.g. ITER. The major concerns are related to the risk of pressure rise and explosion under air or water leak on hot dust, fuel inventory and to the degraded of performance of diagnostic components.

This contribution provides a comprehensive account on: (a) properties of carbon and metal dust formed in the TEXTOR tokamak (b) dust generation associated with removal of fuel and co-deposit from carbon PFC; (c) surface morphology of wall components after different cleaning treatments. Observations of dust during the plasma operation was accomplished by fast cameras and spectroscopy, whereas the material retrieved from the vessel was examined with a number of materail research techniques. The results are following: (i) the amount of loose dust found on the floor of the TEXTOR liner does not exceed 2 grams; (ii) the size varies from $0.1\,\mu$m to 1 mm; (iii) carbon is the main component, but there are also magnetic and non-magnetic metal agglomerates; (iv) the majority of eroded carbon is found in flaking co-deposits on the limiters ($\sim$ 90 g) and neutralizer plates in the pumping ducts ($\sim$ 25 g; (v) the fuel content in dust and co-deposits varies from 10% on the main limiters to 0.03% on the neutralizers; (vi) the presence of fine (up to $1\,\mu$m) crystalline graphite in the collected matter suggests that brittle destruction of carbon PFC could take place during off-normal events; (vii) fuel removal by oxidative methods or by annealing in vacuum disintegrates co-deposits and, in the case of thick layers, makes them brittle and the adherence to the substrate is reduced. Also photonic cleaning by laser pulses produces debris, especially under ablation conditions. The results obtained strongly indicate that in a carbon wall machine the disintegration of flaking co-deposits on PFC is the main source of dust. The issue of metal dust (beryllium and tungsten) formation is still to be properly addressed. It will be done in connection with ITER-Like Wall operation of JET.

EXD/P3-17

Initial Phase Wall Conditioning in KSTAR

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Wall conditioning in KSTAR consists of glow discharge cleaning (GDC), ICRH Wall Conditioning (ICWC), and boronization (Bz). GDC is regularly performed as standard wall cleaning procedure. In order to optimize operation time and removal efficiency, parameter scan has been performed. Another cleaning technique is ICWC, which is useful even though strong magnetic field is present. Thus it is ideal for in-between shot cleaning. To get best operation condition for in-between shot wall conditioning, parameter study for ICWC has been performed in a dedicated session. Bz is a standard technique to remove oxygen impurity from vacuum vessel. KSTAR has utilized carborane powder which is non-toxic boron containing material. A number of tokamaks have been using carborane as boronization material. The first bz has been performed successfully by using 40 g of carborane. Water and oxygen level in vacuum vessel have been reduced significantly. a-C:B:H thin film deposited by carborane is under study at cEps to optimize oxygen gettering capability. In this paper, we will overview the wall conditioning in KSTAR in 2009 campaign.

EXD/P3-18

Plasma Modeling Results, Control Improvement for NSTX and Applications to ITER

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Unlike general control systems, tokamaks have very fast time scales and large unmodeled disturbances, but limited and expensive experimental control-development time. In preparation for ITER, the control tuning and the system-identification methods that fit these constraints must be developed, and incorporated in current tokamaks. In this paper, new control implementations and dynamics studies on NSTX, which further this aim, are summarized. These are, in particular, strike point (SP) position control and vertical stability analysis experiments, and two new system-identification methods / control-tuning

algorithms, implemented on NSTX. The PID controller for the SP was tuned by analyzing the step response of the SP position to the poloidal coil currents, employing the Ziegler-Nichols method. The resulting SP controller was successfully employed to achieve the "snowflake" divertor configuration in NSTX. Controller tuning via experiments can be time intensive. To maximize the proportion of this process that is conducted offline, we implemented an offline system identification of the plasma response to the control inputs based on ARMAX input-output models. With this tool, rough estimates of the improvements were realized and several control improvements were identified. Also, an online automatic relay-feedback PID tuning algorithm was implemented which has the advantage of tuning the controller in one shot, thus optimizing the use of experimental time. Finally, the NTSX contribution to the vertical controllability analysis of ITER is presented where the maximum controllable displacement, ΔZ_{max}, was experimentally obtained and compared with other devices and numerical models. This study improved our understanding of the lack of control capability in the ITER design and resulted in the addition of new inner coil sets to stabilize the vertical mode.

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EXD/P3-19

Be Migration Studies at JET and their Interpretation by an Integrated Model for Plasma Impurity Transport and Wall Composition Dynamics

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[5]*See Appendix of F. Romanelli et al., Nucl. Fusion* **49** *(2009) 104006.*

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The migration of beryllium was studied in JET firstly to provide insight into the underlying transport processes and secondly to establish a reference for comparison to the upcoming bulk Be wall experiments (ILW). The JET wall was prepared in a state far from equilibrium by means of Be evaporation followed by a series of identical plasma discharges in which the relaxation of the wall composition towards the steady state situation was studied by measurement of wall erosion sources using visual range spectroscopy. The Be erosion flux at the midplane main chamber wall initially increased by a factor 20 whereas the corresponding C flux decreased initially only by 50% with respect to the value before Be evaporation. From this observation one can infer that even extended Be evaporation does not lead to a uniform and closed Be layer on top of the carbon wall elements. The subsequent erosion of the Be layer, uncovering of substrate C and mixing with redeposited C occurs on a timescale of 120 s in experiments where standard limiter phase start-up scenarios were used. After this time the Be flux in the limiter phase has already decreased to about a factor 5 above the reference value, whereas the C erosion flux has increased moderately to about 60% of the reference value. The corresponding decay of the Be flux in L-mode plasmas with early X-point formation and correspondingly short limiter phase is slower with a characteristic time constant of 400 s.

To quantitatively interpret the spectroscopic data, which represent only material gross erosion rates and their change over time due to the evolution of respective material fractions at the wall surface, a new integrated model has been developed, which allows describing the evolution of plasma exposed wall surfaces by impurity transport processes in fusion devices with different first wall materials. The experimentally observed time scale of the Be source evolution is already well reproduced by the model for main chamber and inner divertor surfaces, while at the outer divertor the model predicts a factor of $3-4$ lower source magnitude attributed to deficiencies in the impurity transport model, particularly due to topologic constraints of the used computational grid. This has been resolved and respective simulations with extended grid geometry are under way.

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EXD/P3-20

Scaling of Divertor Heat Flux Profile Widths in DIII-D

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Peak steady-state divertor heat flux, $\hat{q}_{div}$, in high confinement burning plasmas remains an open design issue for next generation high power tokamaks such as ITER, largely because of uncertainties in the heat flux profile width, λ_q. Recent experiments in DIII-D featuring divertor heat flux measurements with improved temporal and spatial resolution show that the heat flux profile width in ELMing H-mode single-null discharges, $\lambda_{q,div}$, varies as $1/I_p$ and is nearly independent of power, in agreement with earlier data [1]. The divertor heat flux was calculated from infrared camera measurements using a new high-resolution fast-framing IR camera ($10^7\,\mu$s frame time and 2.5 mm spatial resolution). The frame rate was sufficiently fast to resolve the quiescient heat flux from that due to ELMs. Scans of plasma current and magnetic field were also obtained at constant edge safety factor, q_{95}, to avoid changing the SOL connection length. Preliminary analysis of these data shows the width decreasing linearly with $|B_T|$, presumably due to the changing plasma current and transport. The relation between the divertor heat flux profile width and the midplane SOL plasma parameters is needed for SOL model validation activities and to heat flux scaling studies for tokamak design; SOL models predict $\lambda_{T_e} = (7/2)\lambda_q$. However, our data show that $\lambda_{T_e} \approx$ const even as $\lambda_{q,div}$ varies by nearly a factor of two. We are comparing our results to simulations from the UEDGE 2D fluid code, which accurately includes the SOL magnetic geometry, divertor geometry and neutral recycling, and radiative losses. In these simulations we are focusing on testing the physics by matching the observed trends rather than detailed profiles since the number of free parameters controlling the divertor/SOL conditions greatly exceeds the number of measurements.

[1] C.J. Lasnier, et al., Nucl. Fusion 38 (1998) 1225.

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EXD/P3-21

Performance of Different Tungsten Grades under Transient Thermal Loads

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Plasma facing components in future thermonuclear fusion devices will be subject to intense transient thermal loads due to Edge Localized Modes, plasma disruptions etc. To exclude irreversible damage to the divertor targets, local energy deposition must remain below the damage threshold for the selected wall materials. For monolithic tungsten the formation of a dense crack network has been identified for plasma pulses with power densities above 0.4 GW/m^{-2} and 0.5 ms duration on target plates preheated to 500°C.

The resulting crack pattern depends strongly on the microstructure (grain shape and size) of the used tungsten grade; in addition, alloying elements can have significant impact on the response to intense transient loads and on the mechanical stability. To quantify these effects a number of tungsten grades which have undergone different deformation processes, alloyed tungsten grades, and tungsten coatings have been investigated in comprehensive electron beam studies. In these tests the incident power density, the base temperature of the test coupons, and the number of beam pulses have been varied systematically. Significant differences have been identified for the cracking threshold. In particular, the double forged tungsten, an ultra-high purity tungsten grade and an alloy with 5% tantalum turned out to be the materials with the lowest cracking threshold temperature. To provide an almost complete set of material data including physical and mechanical characteristics, detailed analyses have been performed in close collaboration between FZJ and SCK·CEN.

To predict the performance of W-armoured high heat flux components under ITER-relevant conditions, transient heat load experiments are required with high cycle numbers up to 100 000 or above. To perform these experiments in a reasonable amount of time, the experiments have to be conducted with enhanced repetition rates (e.g. $f = 25$ Hz) on actively cooled targets. Additional steady state heat loads can be applied to guarantee experimental conditions with surface temperatures below or above the ductile to brittle transient temperature. In first electron beam experiments with a medium cycle number $n = 10,000$ surface roughening and crack formation have been observed for absorbed power densities as low as 0.3 GW/m^{-2} under ELM-specific loads of 0.5 ms duration.

EXD/P3-22

Li Experiments on T-11M and T-10 in Support of Steady State Tokamak Concept with Li Closed Loop Circulation

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The concept of a steady state tokamak with the first wall and plasma facing components (PFC) on the basis of the closed loop of liquid Lithium circulation demands the decision of three tasks: Lithium injection to the plasma, Lithium ions collection before their deposition on the vacuum vessel and Li returning from collector to the zone of injection. For practical solution of these problems in T-11M and T-10 tokamak experiments have been applied Li and graphite rail limiters. In this report the general attention has been paid to the investigation of the Lithium collection by different limiters and the studying of Lithium ions behavior close tokamak boundary. The behaviour of Lithium in the SOL area and efficiency of its collection by limiters in T-11M and T-10 tokamaks were investigated by sample-witness analysis and also (T-11M) by use of the mobile graphite probe (limiter) as a recombination target in relation to the stream of Lithium ions. It was measured, that characteristic depth of Lithium penetration in the SOL area of T-11M is about 2 cm and 4 cm in SOL of T-10. That is equal proportional of their major radius R. The calculations by code SOL-DINA, allow to assume, that a collector of Lithium in the form of a ring limiter with width of 5cm could collect in T-11M tokamak (the small radius $a = 20$ cm) up to 90% of the Lithium injected by a rail limiter-emitter. The quantitative analysis of the sample-witnesses located on local rail limiters T-11M showed, that nearby $60 \pm 20\%$ of the Lithium injected during plasma operating of T-11M had been collected by lateral surfaces of T-11M limiters. It confirms a potential opportunity of collection of the main part of the injected Lithium by the limiters-collectors located in the SOL area of steady state tokamak and by that a realization of the closed loop of Lithium circulation near the tokamak first wall and solution by this way the PFC problem of steady state tokamak-reactor.

EXD/P3-23

Fluctuations, ELM Filaments and Turbulent Transport in the SOL at the Outer Midplane of ASDEX Upgrade

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In ASDEX Upgrade turbulent transport in the scrape off layer (SOL) was investigated by electrostatic and electromagnetic reciprocating probes.

In Ohmic discharges density holes dominate the fluctuations inside the separatrix and density blobs in the SOL. The poloidal velocity of blobs and holes was determined by cross correlation of two Langmuir probe pins. The profile shows a strong velocity shear with an abrupt flow reversal just outside the separatrix, which agrees well with the poloidal density fluctuation velocity profile measurement using the Doppler reflectometry technique. This measurement can also give the radial electric field E_r assuming the phase velocity is small compared to the $\mathbf{E} \times \mathbf{B}$ drift velocity. Using the time averaged floating potential and electron temperature (T_e) profiles to determine the plasma potential and E_r, the Doppler E_r profile cannot be reproduced. Significant T_e fluctuations occur in Ohmic discharges which are out of phase with the floating potential fluctuations. Measuring in L-mode floating and plasma potential simultaneously, the fluctuation power spectrum shows a much higher level for the floating potential at frequencies above 30 kHz. Using a ball pen probe (BPP) the plasma potential can be measured directly. The E_r profile derived from the BPP data agrees well with the E_r calculated from the reflectometer data supporting that the poloidal motion of the density fluctuations is governed by $\mathbf{v_E} \times \mathbf{B}$.

T_e of individual ELM filaments was measured using single probes with a fast swept bias voltage. Density and T_e peaks of 10^{19} m^{-3} and $30 - 40$ eV were detected 4.5 cm outside the separatrix. BPP measurements in the SOL showed the ion temperature is higher than T_e. This still has to be validated in ELM filaments. First experiments with a retarding field analyser at ASDEX Upgrade indicate ion energies exceeding 160 eV in the far SOL during ELMs. ELM filaments are thought to be related to current filaments. In previous experiments magnetic fluctuations measured during ELMs could be explained by mode structures in the confined plasma carrying a bidirectional current. New experiments with improved diagnostic were indicating mono polar current filaments in the SOL with current densities of up to 6 MAm^{-2}. Additionally, the small ELMs in type II ELMy and N_2 seeded discharges will be compared to type I ELMs.

EXD/P3-24

Power and Particle Exhaust Control in All W ASDEX Upgrade

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Independent of the plasma facing materials in future fusion devices impurity seeding will become an inevitable element of the operation to protect the divertor from excessive heat loads. Optimisation of the divertor impurity enrichment and the radiation distribution between core, SOL and divertor will be required for devices with high values of P/R (ratio of loss power and major radius). Another important constraint is the fact that the seeding scenario must be integrated into a small ELM regime or combined with ELM mitigation techniques. Impurity seeding has become necessary in the all-tungsten clad ASDEX Upgrade for high power conditions. A very beneficial behaviour in terms of reduced power loads, moderate impurity concentrations and increased confinement has been found in N seeded discharges. Radiative cooling has been applied in a large variety of plasmas ranging from improved H-Modes at intermediate density and heating power to discharges with very high heating power ($P_{aux} \sim 20$ MW) or high density and radiation fraction exploring the type III ELM regime. Generally, the radiated power from the X-point and divertor region increased in N-seeded discharges by more than a factor of two to about 30% of the total input power. There is also a slight increase of radiation from the edge of the main plasma, but the core radiation is almost unchanged. Similarly, also the radiation during type I ELMs is increased. In unseeded discharges with a pure tungsten wall, about 20% of the ELM energy is radiated, which is clearly less than in earlier discharges with mixed carbon and tungsten PFCs. For nitrogen seeded discharges, however, the ELM energy is generally smaller, and about 40% of the ELM energy is radiated. This value is comparable to that found in former campaigns with mixed C/W PFCs and in line with simulations at JET for ELMs with similar energy loss. Consequently, the power load to the divertor targets during and in between type-I ELMs drops significantly with nitrogen seeding. In seeded type III ELMy discharges with good confinement ($H_{98(y,2)} \sim 1$) at $\beta_N \sim 2.3$, power densities below 2 MWm^{-2} in the outer divertor and strongly suppressed W influx could be achieved. Investigations using N_2/Ne or N_2/Ar as seeding gas revealed a change in central transport compared to discharges with N-seeding only.

EXD/P3-25

Impurity Effects on Hydrogen Isotope Retention in Boronized Wall of LHD

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Boronization is one of useful wall conditioning techniques. It is thought that boron will be deposited on the first wall with the impurities and energetic hydrogen isotopes escaped from plasma will be implanted into boron film. Therefore, for the evaluation of hydrogen isotopes recycling in fusion devices, impurity effects on retention behaviors of hydrogen isotopes in boron films were investigated using CVD boron films with controlled impurity concentrations and ones produced by boronization in the Large Helical Device (LHD). It was found that two different existing states of impurities were observed for both boron films. One was the atomic state and the other was that bound to boron chemically. The hydrogen isotopes retention decreased as the total impurity concentration increased, indicating the hydrogen isotopes were desorbed with the formation of heavy water and hydrocarbons. However, some of hydrogen isotopes were retained with much stable chemical forms of O-D and/or C-D bonds with impurities. It was found that these hydrogen isotope trapping and detrapping processes would be influenced by the impurities existing as the atomic state and bounding state with boron, respectively. Furthermore, for the boronized film in LHD, the impurity concentration increased after the exposure to H-H main discharge in LHD, indicating that the impurities in plasma of fusion devices would be dynamically contained into boron film and the removal of hydrogen isotope will largely correlated with that of impurity. It was concluded that these facts will be important to the evaluation for the recycling of hydrogen isotopes in fusion devices.

EXD/P3-26

Fuel Inventory in Carbon Fiber Composites from Tokamaks, Detailed Mapping and Quantification

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The major issues to be tackled in the field of Plasma-wall interaction (PWI) are lifetime of plasma-facing materials (PFM), fuel inventory and dust formation. The aim of this work was to investigate fuel retention and its distribution in a CFC material, NB41 which is an EU reference material for divertor tiles in ITER. Samples of NB41 were exposed to high flux deuterium plasmas in a plasma-wall interaction simulator PISCES-A at the University of California at San Diego, USA, and in the TEXTOR tokamak in Jülich Germany. CFC from Tore Supra at Cadarache France have also been investigated. The

reaction $^2D(^3He,p)^4He$ was used for deuterium and $^{12}C(^3He,p)^{14}N$ for carbon detection using both a broad beam and a nuclear microbeam at the 5 MV Tandem Accelerator in Uppsala.

After the exposure in PISCES-A, two regions with distinctly different deuterium content can be distinguished. They correspond to two different types of fibres, ex-PAN and ex-pitch where the larger roughness of the broken fibres of the ex-pitch region can explain the higher retention in this region.

Micro-distribution of deuterium and carbon in the erosion zone of the target exposed in TEXTOR is shows that the distributions of D and C are not uniform. In complex pattern there are some spots, up to $100-200\ \mu$m wide, with high deuterium content (8×10^{17} cm^{-2}), whereas the remaining part retains less fuel. There are also places with very small relative carbon content, thus indicating the presence of other species accumulated in those areas. Over a distance of less then 1 mm differences in the deuterium concentration was found to be as large as 4% and as low as 0.2% from CFC samples from Tore Supra first wall.

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EXD/P3-27

Results from Radiating Divertor Experiments with RMP ELM Suppression

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The successful integration of ELM suppression using resonant magnetic perturbations (RMPs) with puff-and-pump radiating divertor operation is demonstrated. In addition, because higher gas injection rates are needed to maintain plasma density after the RMP coils have been activated, a radiating divertor with resonant magnetic perturbation (RMP) produces considerably higher levels of radiated power from the divertor and scrape-off layer (SOL)/edge plasma regions than comparable non-RMP discharges at the same density. The radiating divertor has long been proposed as a reliable way to moderate steady heat flux at the divertor targets. Studies at DIII-D have demonstrated that RMPs are effective in suppressing ELMs and thus might be an attractive way to deal with the transient ELM-related heat flux problems expected for ITER. However, it was not clear as to whether RMP-based ELM suppression and radiating divertor scenarios were compatible. Recent DIII-D experiments comprise our first attempts to assess this compatibility by directly comparing the behaviors of injected “seed” argon impurities in RMP and non-RMP puff-and-pump environments and by identifying issues that might limit the use of RMP ELM suppression with a radiating divertor approach. In the puff-and-pump scenarios used, argon was injected in the private flux region of a single-null magnetic configuration near the outer divertor target, while plasma flows into the divertor were enhanced by a combination of particle pumping near the outer divertor target and deuterium gas puffing upstream of the divertor targets. Differences in argon accumulation in the main plasma between RMP ELM-suppressed and similar non-RMP ELMing H-mode plasmas were relatively small, typically less than 20%. The core concentration of argon decreased as the deuterium gas puff rate was raised in RMP and non-RMP cases, suggesting that the detailed UEDGE analysis reported previously [1] for non-RMP divertor and SOL radiating divertor plasmas could also be valid for RMP cases. As the deuterium and argon gas puff rates were increased, the density and the edge electron pressure gradient also increased, leading ultimately to the re-appearance of the Type-1 ELMing activity.

[1] T.W. Petrie, et al., Nucl. Fusion **49** (2009) 065013.

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EXD/P3-28

Dynamic Wall Loads Measured by Gas Balance Technique in all Tungsten ASDEX Upgrade

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In vessel tritium inventory is still a safety issue for ITER. The amount and mechanism of hydrogen retention depends on the plasma facing component (PFC) material. During the last years AUG demonstrated that the long term deposition of D is strongly reduced during the transition from carbon to full tungsten PFCs. However deposition measurements yield no time resolved information, which is desired to verify predictions to ITER. In contrast time resolved gas balance techniques allow to identify the mechanism relevant for the retention. According to the gas balance measurements, high density H-mode discharges can be divided into different phases. First, after plasma start up, the PFCs strongly retain D, until the wall is saturated. Then, for all W PFC device, a steady state phase is reached, during which wall pumping and release are equal within the error bars. Outgassing starts immediately when the gas puffing rate is reduced or the discharge ends. Obviously, the in-vessel D inventory is dominated by dynamic wall loads. For W PFCs the long term retention in AUG is below a few percent of the amount of gas puffed. During the 2009 campaign AUG was operated again with pure tungsten PFC, i.e. boron layers had been completely removed during the vent. During the 2008 campaign the amount of gas, which is temporally retained before reaching wall saturation, was 2×10^{22} at for the unboronized and 1.5×10^{22} at for the boronized wall. In 2009 only $1.2 - 1.5 \times 10^{22}$ at are needed, even if the wall was not boronized. To increase further the accuracy of the measurements discharges were performed using the cryo-pump only, as proposed by Alcator C-mod. In AUG this procedure only allows operation with low gas puffing levels. High medium term retention rates of 25 to 50% of the amount of gas puffed were observed in these kind of experiments, comparable to Alcator C-Mod results. But the absolute amount of retained gas agrees well with the value observed for high density discharges. Therefore the earlier apparent discrepancy of the C-mod and AUG results seems to be due to the different amount of gas puffed during the discharges. The total amount of the dynamically retained gas and its temporal behavior will be compared to the ion and neutral fluxes to the wall.

EXD/P3-29

Magnetic Structures and Pressure Profiles in the Plasma Boundary of RFX-mod: High Current and Density Limit in Helical Regimes

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New edge diagnostics and detailed analysis of the magnetic pattern have significantly improved the comprehension of the processes that develop at the boundary of the Reversed Field Pinch (RFP) plasma in RFX-mod ($a = 0.46$ m, $R = 2$ m). An upper critical density threshold $n_C = 0.4\, n_G$ (n_G Greenwald density) is seen to limit the operational space for the improved Quasi Single Helical (QSH) regime: magnetic topology reconstruction and diagnostic observations suggest this limit as due to helical plasma wall interaction (PWI) in combination with toroidally localized edge density accumulation and cooling. The experimental evidence is provided by a reconstruction of the magnetic topology, the space and time resolved pattern of the floating potential measured at the wall, the plasma shift induced by the dominant ($n = -7$) toroidal mode, the electron pressure profiles measured by a Thermal Helium Beam (THB) diagnostic and the toroidal plasma flow obtained from Gas Puffing Imaging (GPI) system. In QSH regimes the toroidal modulation of electron pressure, floating potential and radial shift of the dominant mode are mainly determined by a helical PWI, with a source term localized where the dominant mode shift points outwards and a possible stagnation point at the X-points of $m = 0$ island chain, which is present in the plasma boundary. Such interaction increases with plasma current. The equilibrium between particle source and sink (which depends on wall recycling) does affect the possibility of having a QSH-Multi Helical state transition: the QSH state cannot be maintained beyond a critical density $n_C = 0.4\, n_G$. The possibility to increase n_C with proper wall conditioning (e.g. lithization), which would decouple the fluxes from the wall and from the reversal region is under experimental verification. This density limit in the RFP plasma that limits the operational space where QSH is produced has remarkable similarities in the Tokamak H-mode physics whereby the Low to Henhanced (L-H) mode transition may be obtained below an upper density limit known as the Borrass parameter $n_{BLS} = 0.4\, n_G$: in this case this behavior has been imputed to anomalous density accumulation in the divertor X-point. The topology of RFP plasma boundary is even richer at different equilibrium (deeper reversal in the edge) where resistive interchange g-modes have also been observed.

EXD/P3-30

Key Results from the DIII-D/TEXTOR Collaboration on the Physics of Stochastic Boundaries projected to ELM Control at ITER

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In this paper key results of a direct comparison of TEXTOR and DIII-D experiments with external resonant magnetic perturbation (RMP) fields are presented. These comparisons of resistive, low rotation L-mode plasmas at TEXTOR to highly conductive, high rotation H-mode plasmas at DIII-D identify generic physics mechanisms of the application of RMP fields with a strong field line pitch angle alignment in the plasma edge. We show evidence that this pitch alignment induces a stochastic edge layer in both very different plasma regimes with at least a thin layer of open field lines in the plasma edge breaking the axis-symmetry of the tokamak introducing a new three-dimensional (3D) plasma boundary. In both plasma regimes a reduction of the electron pressure p_e with increasing extension of the modeled stochastic layer and p_e recovery with decreasing stochastic layer width is found. This $p_e(q_{95})$ resonant modulation is caused consistently by a strongly q_{95} resonant reduction of the electron temperature correlated to the stochastic layer width. This compares well to the generic physics picture of an open stochastic layer and the p_e reduction induced is identified as a pre-requisite for stabilizing type-I ELMs by RMP. The connection between the magnetic topology and plasma transport was studied with the EMC3-Eirene plasma and neutral transport code and revealed a comparable impact on both devices of the perturbed layer on the plasma profiles and the plasma wall interaction. The potential to reduce the RMP induced particle pump out by fine tuning the RMP spectral properties was demonstrated experimentally. At low resonant field amplitudes enhanced particle confinement is shown in high field side limited L-mode discharges on both devices while higher resonant field amplitude yield particle pump out. This shows the potential for optimizing ELM control by RMP towards maintaining high plasma density and proves the importance of taking into account the 3D plasma boundary induced during ELM suppression for design of first wall components at ITER.

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EXD/P3-31

Pedestal Characterization and Stability of Small-ELM Regimes in NSTX

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An instability near the plasma edge known as the edge harmonic oscillation (EHO) is thought to enable access to the ELM-free QH-mode in tokamaks, which is a highly desirable operational regime for ITER because of the avoidance of periodic ELM heat loads. The EHO has been hypothesized to be a saturated kink driven unstable by toroidal rotational shear that provides sufficient transport near the plasma edge to keep the edge pressure below the peeling-ballooning stability limit. NSTX has observed unstable modes with similar characteristics to the EHO coincident with transition to a small-ELM regime. These small ELMs do not have a measurable effect on the plasma stored energy ($\ll 1\%$). Transition to this regime is associated with a downward biased plasma as evidenced by drsep < -5 mm. Soft x-ray emission indicates that these modes are localized just inside the pedestal and are correlated with increased density fluctuations in the pedestal as measured by microwave reflectometry. The lowest order mode rotates at the plasma rotation frequency, indicating $n = 1$, and harmonics up to $n = 3$ have been observed simultaneously with the $n = 1$, as determined by the rotation frequency of the higher harmonics. Stability analysis during the observed modes indicates instability to $n = 1-3$ with $n = 3$ having the highest growth rate and unstable mode eigenfunctions peaked near the plasma edge. Rotational shear has been shown to shift the most unstable mode to lower-n, possibly explaining the observation of $n = 1$ being the predominant mode. Significant rotational shear has been observed at the mode location in these small ELM discharges, consistent with the rotational destabilization of the EHO. In contrast, discharges with similar shape and pedestal pressure without the edge instability and with large type-I ELMs have reduced rotational shear at the pedestal.

EXD/P3-32

Synergy between the Innovative "Snowflake" Divertor Configuration and Lithium Plasma-Facing Component Coatings in NSTX

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The studies of an innovative "snowflake" divertor configuration and evaporated lithium wall and divertor coatings in the National Spherical Torus Experiment (NSTX) provide support to these PMI concepts as viable candidates for future high divertor heat flux tokamaks and spherical tokamak based devices for fusion development applications. Lithium coatings have enabled ion density reduction up to 50% in NSTX through the reduction of wall and divertor recycling rates. The outer SOL transport regime changed from the high-recycling, heat flux conduction-limited with $\nu_e^* \sim 10-40$ to the sheath-limited regime with a small parallel T_e gradient and higher SOL T_e with $\nu_e^* \leq 5-10$. A recycling coefficient of $R \sim 0.85$ was inferred from interpretive two dimensional multifluid edge transport modeling. However, a concomitant elimination of ELMs and an improvement in particle confinement caused impurity accumulations. The "snowflake" divertor (SFD) configuration obtained in NSTX in 1 MA $4-6$ MW NBI-heated H-mode lithium-assisted discharges demonstrated encouraging impurity control and divertor heat flux handling results. A number of theoretically predicted geometric and radiative properties of the SFD configuration has been confirmed. A very high poloidal flux (area) expansion of the separatrix region in the SFD, as well as a longer connection length, as compared to a standard divertor configuration, led to a partial strike point detachment and the associated peak heat flux reduction. The core carbon density and radiated power were also significantly reduced.

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EXD/P3-33

Inhibition of C: H Co-deposit Formation by Ammonia Injection in Remote Areas of ITER

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Since its proposal in 2002 the scavenger technique for the inhibition of tritium rich co-deposit formation in hidden areas of carbon-based Fusion Reactor has been tested in several tokamak devices, divertor simulators and laboratory cold plasmas. For nitrogen injection, however, the concern about a possible enhancement of carbon erosion by the injected scavenger species at the divertor plasma still exists, so that alternative injection schemes must be developed. In the experiments here reported, ammonia has been tested as a possible candidate for the use of the scavenger concept in a non-perturbative way i.e., injected away from the plasma-wall interaction area.

In Pilot-PSI, injection of nitrogen and ammonia in an H_2/methane plasmas (2.0 and 0.16 slm respectively, total pressure of 4 Pa) was performed at two different locations with respect to the deposition monitor, located $\sim$ 15 cm downstream the target and screened from the plasma column. In order to enhance the transport of the plasma-borne carbon radicals, a hot liner (300°C) was installed coaxially to the plasma column. In the absence of scavenger, a carbon deposit, whose thickness was monitored in situ by laser interferometry, was created at a rate of 4 nm/min. Two locations of the gas inlet were tested, feeding the scavenger species either directly into the plasma column or a few centimetres in front of the deposition monitor, in the plasma-free area. Optical emission spectroscopy (OES) was used to record the impact of the seeded gas into the plasma. Although inhibition factors $>$ 80% were achieved for both species, it was found that ammonia not only works more efficiently as carbon film scavenger than nitrogen does but, also, its efficiency as scavenger is basically independent of the location respect to the plasma where it is seeded.

Furthermore, Inductively Coupled RF plasmas in ammonia/methane were used to test the scavenging effect of ammonia in the absence of ionic species. The results point to the presence of a fast reaction between ammonia and the carbon radicals created in the methane plasma. In this work, these results will be presented and their implication on the strategy for hydrogen retention control by the scavenger technique in the non-activated phase of ITER will be addressed.

EXD/P3-34

Towards a Comprehensive Approach of Edge and SOL Transport Issues: from Experimental Results to Global Simulations

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Bringing answers to the questions still open regarding edge transport issues in ITER requires a comprehensive approach bridging experiments to simulations. We report here about such an effort. We first present results from the Tore Supra (TS) tokamak. Experiments were led to study the dynamics of edge turbulence by correlating various diagnostics. Data evidence the existence of structures aligned on the magnetic field, localized on the low field side with similar dynamics to that observed on other devices. Filaments were also observed in the closed field lines region during detached discharges. Such results obtained in a limiter machine demonstrate that the phenomenology of the edge turbulent transport is little sensitive on the magnetic topology. Another set of experiments was devoted to the measurement of the ion temperature in the edge plasma. Large T_i/T_e ratios are found in the Scrape-Off Layer (SOL). In highly radiating scenarios, T_i drops more than T_e implying that the energy dissipated through electron radiation losses is supplied mainly via the ions. To back these results, a theory and modelling effort, the ESPOIR project, has been initiated. Global transport aspects have been studied with the SOLEDGE-2D fluid code. In particular, simulations allow us to recover regimes with $T_i/T_e > 1$ in the SOL, T_i being more sensitive than T_e to an enhanced volumetric sink on the electron channel. In parallel, more fundamental work has been undertaken on the numerical treatment of edge boundary condition leading to a novel method based on penalization techniques. The method offers greater flexibility in the configuration of plasma facing components and allows the use of powerful numerical algorithms. The 3D dynamics of filaments is also analysed with an analytical model based on front propagation that is confronted to Tore Supra data. Finally, first results of a 3D turbulent code (SOLEDGE-3D) are presented. The occurrence of the Kelvin-Helmholtz instability in self-consistent equilibriums is analyzed. An ITER geometry version is also being developed and first simulations dedicated to radiative layers and the associated fronts and instabilities are presented. To conclude, we suggest defining the Mistral Base case from the TS data base. This would provide a means to compare edge fluid and kinetic simulations as achieved with the Cyclone base case for core turbulence.

EXD/P3-35

Gross and Net Chemical Erosion of Carbon at High Fluxes of Low Temperature Hydrogen Plasma

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Carbon can withstand extreme heat loads and is therefore considered for ITER as a wall material for the areas with the strongest particle and power loads. The main disadvantage of carbon is its chemical reactivity with hydrogen ions and atoms. Consequently, erosion will occur even at very low plasma temperatures, compromising not only the life time of the carbon target but also introducing secondary effects as hydrogen retention by deposition of hydrocarbons and dust formation by flaking off of such layers.

The linear plasma generator Pilot-PSI was used to investigate the chemical erosion of carbon at ITER divertor relevant plasma flux densities ($\sim 1 \times 10^{24}$ /m^{2}s) and plasma temperatures at which only chemical processes are significant (~ 1 eV). The erosion of fine grain graphite was in situ measured with optical emission spectroscopy on the A-X transition of the CH radical. These data were complemented with ex situ post mortem surface profilometry and confocal microscopy measurements to compare the instantaneous eroded particle fluxes with the net amount of material that were removed. The measurements were performed in scans of the plasma flux density between 2×10^{23} and 4×10^{24} /m^{2}s and of the plasma temperature between 0.3 and 2 eV.

In this contribution we report on the experimental results. These show that the chemical erosion increases by a factor of ~ 7 over the plasma temperature range between 0.3 and 1 eV and saturates to 6×10^{22}/m^{2}s at temperatures above 1 eV. Taking this plasma temperature effect into account learns that the chemical erosion does not depend on the plasma flux density, which implies that the chemical erosion yield is inversely proportional to the plasma flux density. Finally it is shown how subtle changes in the plasma temperature are sufficient to change the area of the peak plasma flux density from a net erosion zone into a net deposition zone. These net deposits are observed to act as a protective layer against chemical erosion for the underlying surface.

EXD/P3-36

Main Chamber Plasma-Wall Interaction Studies in DIII-D in Support of ITER

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Recent experiments in DIII-D examined heat and particle flux profiles to main chamber regions that are of particular concern for the design of ITER's first wall. The first of these are at the midplane where ITER is expected to limit plasmas during the rampup and rampdown phases of discharge. The scrapeoff layer (SOL) power flux density e-folding length, λ_q, is a critical design parameter for limiters. Scans of the plasma current, density, and input power were performed in DIII-D to test the modified divertor L-mode scaling of λ_q used for ITER. During inner wall limited (IWL) L-mode discharges λ_q is on average ~ 2.5 times larger than in diverted and upper outer wall limited plasmas, in agreement with the assumptions used in the ITER limiter design. The measured values of λ_q are in reasonable agreement with ITER scaling, constituting the first direct test of this scaling in limited plasmas and confirming assumed strong dependence of λ_q on the device size. However, scaling dependencies of λ_q on individual discharge parameters could not be confirmed. The other concern for main chamber interactions in ITER is at the top of the vessel where a secondary divertor is expected to handle both steady state and edge localized mode (ELM) levels of particles and energy. The IRTV-measured secondary divertor heat flux exhibits an overall decrease for smaller ELM regimes (higher density) and magnetic configurations with a greater distance (drsep) between the primary and secondary separatrices. During ELMs, fast IRTV line scan shows a multi-peaked surface temperature profile and IRTV image data shows multi-peaked spiral patterns. Steady-state IRTV heat flux patterns show an exponential decay length, λ_q, near the secondary strike point similar to thermocouple time averaged heat flux profiles and 4 times larger than typical primary divertor cases. Thermocouple heat flux profiles were determined from fast response thermocouples placed 1 cm deep in the main graphite divertor tiles. Scans were performed in density and drsep for 2 plasma shapes with different triangularity. Secondary divertor carbon deposition was measured though surface analysis of tiles following C^{13} marked methane gas injection experiments with a secondary divertor.

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EXD/P3-37

ICRF Impurity Behavior with Boron Coated Molybdenum Tiles in Alcator C-Mod

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Although ion cyclotron range of frequency (ICRF) heating is considered an excellent candidate for bulk heating, minimizing impurity production associated with ICRF operation, particularly with metallic plasma facing components (PFC), remains one of the primary challenges for ICRF utilization. In C-Mod and present experiments, boronization, an insitu applied boron film, is utilized to control impurities and its effectiveness has a limited lifetime. In C-Mod, the lifetime has been observed to be proportional to integrated injected RF Joules and the degradation is faster than in equivalent ohmic heated discharges the ICRF is enhancing the erosion rate of the boron film [1].

In an effort to identify important erosion and impurity source locations, we have vacuum plasma sprayed $\sim$ 100 microns of boron on molybdenum tiles from the outer divertor shelf, main plasma limiters, and the RF antennas. We have also modified the shape of the main plasma limiter and increased our spectroscopic monitoring diagnostics of the main plasma limiter. Finally, we have installed a set of probes to monitor the plasma potential and RF fields on field lines connected an antenna.

For ICRF heated H-modes, the core molybdenum levels was significantly reduced and remained at low levels for increased integrated injected RF Joules. The core molybdenum levels also no longer scales with RF power in L-mode in contrast with previous results with boronization and molybdenum plasma facing components [2]. Initial Post campaign analysis of the boron coating will also be presented.

Boronization and impurity, typically nitrogen or neon, seeded discharges enabled high plasma and ICRF antenna performance. The boronization suggests that other impurity sources are important but are yet to be identified. Impurity seeding had two important effects: reduced core molybdenum levels and suppressed antenna faults due to arcs and injections from antenna structure. The lower core molybdenum level is surprising since one may expect the sputtering to increase in the presence of a medium-Z, recycling impurity. Results from these experiments will be presented.

[1] B. Lipschultz et al., Phys. Plasmas 13, 056117 (2006).

[2] B. Lipschultz et al., Nuclear Fusion 41, 585 (2001).

EXS — Oral

(*Magnetic Confinement Experiments: Stability*)

EXS/1-2

Advances toward QH-mode Viability for ELM-Free Operation in ITER

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The application of static, non-axisymmetric, nonresonant magnetic fields (NRMFs) to high beta DIII-D plasmas has allowed sustained operation with a quiescent H-mode (QH-mode) edge and both toroidal rotation and neutral beam injected torque near zero. Previous studies have shown that QH-mode operation can be accessed only if sufficient radial shear in the plasma flow is produced near the plasma edge [1]. In past experiments, this flow shear was produced using neutral beam injection (NBI) to provide toroidal torque. In recent experiments, this torque was nearly completely replaced by the torque from applied NRMFs. These results open a path toward QH-mode utilization as an ELM-free H-mode in the self-heated burning plasma scenario, where toroidal momentum input from NBI may be small or absent.

The application of the NRMFs does not degrade the global energy confinement of the plasma. Conversely, it increases plasma resilience to locked modes and allows plasma operation at very low NBI torque and core plasma rotation. According to neoclassical theory [2], at low rotation the NRMF torque is amplified by entering a regime where the radial electric field vanishes, leading to "superbanana" particles with enhanced radial fluxes, which in turn exert a larger toroidal torque on the plasma. Measurements of the dependence of this torque on the plasma velocity have shown that a peak in the torque exists at very low plasma rotation, and that this peak is found to occur where the radial electric field is small, as predicted by theory. Experimental results also show that surprisingly, the energy confinement quality increases at low plasma rotation. Furthermore, the beneficial effects of the applied NRMF are enhanced as the energy content of the plasma increases, likely because of plasma amplification of the applied NRMF. All these results project favorably toward the realization of an ELM-free "quiescent" self-heated burning plasmas in ITER, where the torque from NBI is expected to be small.

[1] K.H. Burrell et al., Phys. Rev. Lett. **102**, 155003 (2009).

[2] K.C. Shaing et al., Plasma Phys. and Control. Fusion **51**, 035009 (2009).

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EXS/5-1

Effect of Magnetic Materials on In-Vessel Magnetic Configuration in KSTAR

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It has been observed in KSTAR that there is a significant discrepancy between the measured and the calculated poloidal vacuum magnetic field (B). It is mainly due to the presence of an inherent magnetic material used for the conduit of the superconducting strands. It was reported in the last two campaigns that the operation boundary of the startup process in KSTAR was narrow both in pre-fill neutral pressure and field-null size and hence the discharge reproducibility was poor, for which the additional field from Incoloy might be responsible together with the wall conditioning issue. From the initial analysis of Incoloy, it seems that the effect of Incoloy on the startup will not be trivial and it will degrade the initial field-null quality but the position stability of the initial plasma column will be affected by the modified field index. In this study, firstly, the FEM model will be thoroughly benchmarked and validated further with the in-situ measurements. The detailed measurements at the field-null region with the above systems will be presented and the additional field from Incoloy will be clarified. Based on a reliable FEM model of Incoloy, a new optimization toolkit for startup scenarios will be developed taking into account the non-linear effect of the PF Incoloy. Using this technique, the degradation of field-null quality will be systematically compensated in terms of the field-null size and the position stability. The developed scenario will be tested experimentally during the KSTAR 2010 campaign and its impact will be discussed on the operational boundary and reproducibility of plasma.

EXS/5-2

Evidence of Stochastic Region near a Rational Surface in Core Plasmas of LHD

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Clear evidence of stochastization of the magnetic surfaces near a rational surface is observed in the core plasma with weak magnetic shear in the Large Helical Device (LHD) by applying heat pulses driven by modulated electron cyclotron heating (MECH). The stochastization of the magnetic surfaces is confirmed by the observation of 1) flattening of electron temperature profiles, 2) very fast propagation of the heat pulse, which is in contrast to the slow heat pulse propagation observed in the electron temperature flat region of the nested magnetic island.

One piece of evidence of stochastization of the magnetic surfaces is the flattening of T_e profile[1], however, the flattening can occur due to the lack of heat flux crossing the magnetic flux as seen in the magnetic island even if the magnetic flux surface is nested inside.

In this paper, experimental confirmation of stochastization of the magnetic surface is carried out by the analysis of heat pulse propagation driven by modulated electron cyclotron heating (MECH) with a frequency of 39 Hz and cold pulse propagation driven by TESPEL in the plasma with a flat electron temperature profile, which is observed when the magnetic shear at the rational surface measured with motional stark effect (MSE) spectroscopy drops below a critical value ($0.2 \sim 0.5$). When the direction of the NBI is switched from balanced to counter to equivalent plasma current, the magnetic shear at a rational surface at $\iota = 0.5$ decreases, because the rotational transform near the plasma edge decreases due to non-inductive current drive by NBI and the central rotation transform even increases transiently due to the inductive current. When the magnetic shear drops below the critical values, the large core flattening of electron temperature up to half of plasma minor radius and partial flattening at one-third of plasma minor radius are observed. Since the power deposition of MECH is localized at the magnetic axis, a heat pulse propagates from the plasma center toward the plasma edge (outward) when there is no magnetic island. However, the heat pulse propagates rapidly in the electron temperature flat region (no time delay within the accuracy of the measurements), which is a clear evidence stochastization of magnetic surface.

[1] K. Ida, et al., Phys. Rev. Lett. 100 (2008) 045003.

EXS/5-3

Interactions between MHD Instabilities in the Wall-stabilized High-β Plasmas

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In the JT-60U wall-stabilized high-β_N plasmas, some interactions between MHD instabilities have been observed. First one is an energetic particle driven wall mode (EWM) triggered edge localized mode (ELM). When the EWM appeared, the ELM was synchronized with the EWM, suggesting that the EWM can affect a pedestal stability. Second one is an ELM excited resistive wall mode (RWM). In the high-β_N plasmas where the RWM is marginally stabilized by the plasma rotation, a response related to a stable RWM has been observed. The response is considered to be excited by the ELM impact as an impulse response. The response indicates the margin of the RWM stability.

EXS/5-4

Non-ideal Modifications of 3D Equilibrium and Resistive Wall Mode Stability Models in DIII-D

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Recent DIII-D experiments deliver strong evidence for the applicability of kinetic theory [1,2] to describe resistive wall mode (RWM) stability and the closely related problem of plasma equilibria in the presence of external 3D magnetic fields. These modifications of ideal MHD are the physics basis for passive RWM stabilization, which is very attractive for a fusion reactor since it increases the attainable plasma pressure and thereby fusion performance without the need for a complex magnetic feedback system. Deviations from ideal MHD already become apparent in measurements of the plasma response to externally applied $n = 1$ fields at plasma pressures that are 20% below the ideal MHD no-wall stability limit. The observed stable operation at plasma pressures above the no-wall limit over a wide range of rotation profiles can be explained by kinetic theory, when taking into account energetic ions, which originate from the neutral beam injection (NBI) heating and constitute up to 35% of the kinetic energy of the plasma, in addition to the thermal particles. Measurements of the plasma response to externally applied $n = 1$ fields in plasmas above the ideal MHD no-wall stability limit yield further more direct evidence for the importance of the kinetic wave-particle resonances. Varying the NBI torque in otherwise similar plasmas yields the largest plasma response, indicating the weakest damping of the RWM, when the rotation evaluated at the $q = 2$ surface is approximately 1% of the Alfvén velocity. The weakest damping, thereby, occurs in the rotation gap between the slower

precession frequency and the faster bounce frequency of the trapped thermal particles consistent with kinetic modeling. The present DIII-D experiments are an important step towards a quantitative understanding of wall-stabilization, which is essential in order to develop confidence in predictions for reactor scenarios that rely on wall-stabilization, such as the steady-state scenario in ITER.

[1] A. Bondeson, M.S. Chu, Phys. Plasmas **3**, 3013 (1996)

[2] B. Hu, R. Betti, Phys. Rev. Lett. **93**, 105002 (2004)

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EXS/5-5

Resistive Wall Mode Stabilization and Plasma Rotation Damping Considerations for Maintaining High Beta Plasma Discharges in NSTX

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Maintaining steady fusion power output at high plasma beta is an important goal for future burning plasmas such as in ITER advanced scenario operation and the Fusion Nuclear Science Facility. Research on the National Spherical Torus Experiment (NSTX) is investigating the stability physics and control to maintain steady high plasma normalized beta greater than 5 with minimal fluctuation. Resistive wall mode (RWM) instability is observed at relatively high rotation levels. Analysis including kinetic dissipation effects using the MISK code shows a region of reduced stability for marginally stable experimental plasmas caused by the rotation profile falling between stabilizing ion precession drift and bounce resonances. Collisionality alters the dependence of RWM stability on rotation. Energetic particle (EP) effects are stabilizing and weaker than in tokamaks due to a reduced EP population in the outer plasma. Calculations for ITER show that alpha particles are required to stabilize the RWM at anticipated rotation levels for normalized beta of 3. Non-resonant braking by applied 3D fields could be used to actuate rotation control and avoid rotation profiles unfavorable for RWM stability discussed above. Recent experiments have varied the ratio of ion collisionality to $\mathbf{E}\times\mathbf{B}$ frequency. As the $\mathbf{E}\times\mathbf{B}$ frequency is reduced, the NTV torque is expected to increase as collisionality decreases, and maximize when it falls below the grad(B) drift frequency (superbanana plateau regime). Increased non-resonant braking was observed at constant applied field and normalized beta in experiments when rotation and $\mathbf{E}\times\mathbf{B}$ frequency were reduced to low values (most applicable to ITER) as expected by theory. The newly-developed multi-mode VALEN code is used to analyze high normalized beta experiments showing evidence of driven RWM activity. The computed RWM growth rate vs. normalized beta is in the observed experimental range. The computed spectrum of eigenfunctions comprising the perturbed

field shows significant multi-mode content. Using this model, multi-mode RWM stability is determined for ITER advanced scenario plasmas at normalized beta sufficient to destabilize $n = 2$ modes. Combined RWM and new beta feedback control capability were used to generate high pulse-averaged normalized beta with low fluctuation. NTV braking was used to alter plasma rotation compatibly with beta feedback.

EXS/8-3

Characteristics and Control of Type I ELM in JT-60U

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In ITER, the acceptable ELM energy loss of ~ 1 MJ requires an increase in the ELM frequency by ~ 20 times higher than the natural type I ELM frequency. Since the value relies on the edge collisionality dependence of the ELM amplitude normalized to the pedestal stored energy, the requirement depends on scalings of the normalized ELM amplitude. In JT-60U, a new parameter of the ratio of the pressure gradient inside pedestal to the pressure gradient within the pedestal has been confirmed as an important parameter determining the normalized ELM amplitude. Before the collapse phase in which the pedestal stored energy is expelled, coherent density and temperature precursors are observed. The growth rate of precursor is evaluated to be 10^{-3} times Alfven frequency for the first time, suggesting that the instability with such a small growth rate determines the onset condition of type I ELM. When the plasma at the top of the pedestal at high-field side was heated by 1.35 MW of EC wave, the ELM energy loss was reduced by $\sim 30\%$, from 22 kJ to 16 kJ. Thus, the localized EC injection to the pedestal can be considered as a candidate for an active ELM control technique in ITER.

EXS/10-1Ra

Comparative Study of Sawtooth Physics and Alfvén Waves via 2D ECE Imaging on KSTAR, DIII-D and TEXTOR

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A new and improved 2D microwave imaging system on TEXTOR tokamak plasma demonstrated the unprecedented advantage of high temporal and spatial resolution of 2D images over the conventional 1D data in studies of the physics of the sawtooth crash; The major new findings are; a) The high field side sawtooth crash was re-confirmed. b) The role of the core current density in the theoretical explosive growth rate of the ideal kink or resistive instability may have to be considered to explain the observed multiple reconnection processes and stagnated growth rate of the island during the post-cursor phase of the sawtooth oscillation. (c) The rigid body rotation assumption is well valid and this validation was achieved through variation of the toroidal rotation speed controlled by the toroidal momentum input by the heating beam. Following the successful findings of new physics from the TEXTOR experiments, a state-of-the-art optics design for KSTAR and DIII-D was launched for detailed study of physics. The DIII-D system has already been deployed to study the physics of sawteeth as well as of other MHD modes. The sawtooth results will be compared with the previous studies from TEXTOR where the plasma shape is circular. The first 2D structure of Alfvén eigenmodes from the reverse shear regime in DIII-D discharge is a significant preliminary outcome; the results from the 2010 KSTAR campaign will be compared with those from DIII-D as well as TEXTOR. Abundant spatial ($\sim$ 200 channels), temporal, and frequency information will be utilized to study the fine structure of these modes. The KSTAR system featuring an optical system with a high zoom capability of up to a factor of 3 will be deployed for the 2010 campaign. This system is equipped with twin imaging arrays for simultaneous observation of two different plasma regions and will be used for comparative study of sawtooth physics, Alfvén waves, NTMs and ELMs associated with the L/H transition. One of the prime physics goals of the KSTAR system is to explore the fundamentals of the disruption physics in a superconducting tokamak device. The global as well as the detailed local images of the thermal quenching phase of the disruption will assist the construction of a full picture of disruption physics and may provide a preventive measure of this event which is one of the most important issues for ITER as well as DEMO.

EXS/10-1Rb

Study of m/n = 1/1 and High-Order Harmonic Modes during the Sawtooth Oscillation via 2D ECEI in a Low β Tokamak Plasma

Xu, X.Y.[1], Xu, M.[1], Wang, J.[1], Wen, Y.Z.[1], Yu, Y.[1], Yu, C.X.[1], Wan, B.N.[2], Gao, X.[2], Luhmann, N.C.Jr.[3], Domier, C.W.[3], Xia, Z.G.[3], and Shen, Z.W.[3]

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Sawtooth oscillations were investigated using an electron cyclotron emission imaging (ECEI) diagnostic technique on the HT-7 tokamak. High-order harmonic modes in sawtooth precursors, cause sharp pressure points, leading to the occurrence of reconnection events at more than one place, and which are not preferential on the low field side of the $q \sim 1$ radius at low density [1]. The ECEI observation of the harmonics and sawtooth crash at different current and density is summarized in further. A further study of the modes, indicate that there is a relationship of the 1/1 mode and 3/1 mode at the edge. The evolution and the parametric study of the harmonics is investigated in further. A new optic system designed for the 2D ECEI system on the EAST tokamak(upgrading of the HT-7 tokamak) is described, one of whose purposes is to investigate the the sawtooth oscillation.

[1] Xiaoyuan Xu et al., Plasma Phys. Control. Fusion 52 (2010) 015008.

EXS/10-2Rb

Control of the Nonthermal Electrons and Current Collapse at the Density Limit Disruption in T-10 Tokamak

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Safe termination of plasma discharge is one of the key problems in design of the plasma-facing components in tokamak reactor due to strong electromagnetic forces and heat fluxes arising during collapse of the plasma current and subsequent generation of the runaway electron beams during disruption instability. Present experiments in the T-10 tokamak has demonstrated possibility of the control of runaway electron generation and current decay rate after an energy quench at the density limit disruption using ECRH heating in combination with preprogrammed Ohmic power supply system and gas puffing. Essential new result of the present experiments is identification of the optimal conditions of safe termination of the plasma discharge using auxiliary heating and identification of the nonthermal x-ray perturbations at the initial stage of the current collapse using 2D CdTe tomographic arrays. Analysis has indicated that current decay in the T-10 tokamak is represented by series of minor disruptions accompanied by intensive bursts of the nonthermal x-ray and bolometric radiation and periodic "humps" of the soft x-ray intensity. Rate of the plasma current decay correlates with repetition rate of the bursts which indicates indirectly that current collapse in T-10 plasma is determined by the step-like process associated with the "connection events" inside the plasma core. Auxiliary plasma heating using electron cyclotron waves are used in T-10 to control the MHD modes and nontermal x-ray bursts at various stages of the disruption. Experiments have demonstrated that elimination of the bursting energy quenches (possibly connected with the reconnection) and formation of the saturated MHD modes can reduce the current decay rate, delay formation of the primarily runaway beams, and in some cases assures restoration of the quasi-stable plasma conditions.

EXS/10-3

JET Disruption Studies in Support of ITER

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[6]*See Appendix of F. Romanelli et al., Nucl. Fusion* **49** *(2009) 104006.*

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Disruptions are a key issue for all future large tokamaks, due to the large mechanical and thermal loads they can place on the tokamak structure. Given that size (and field and current) are a key determinant of disruptions loads, JET is best placed to understand ITER disruption issues. This paper summarises recent key advances in understanding on JET, in the disruption area. Results will be presented on halo currents and asymmetric disruptions, where large sideways forces may occur (up to 4 MN) – which are very significant if simply extrapolated to ITER. Considerable effort is being devoted to understanding this asymmetry, its rotation and how it should be extrapolated to ITER. A second significant consequence of disruptions is the resulting thermal loads. IR data show in general that the outer limiter and the divertor each handle comparable energies. A third disruption consequence is runaway electrons (REs). IR imaging shows distinct localised impacts on the upper dump plate by the RE beam leading to an increase of the surface temperature of up to ~ 1500°C. The effects of applied non-axisymmetric $n = 1$ or $n = 2$ fields on the runaway beam have been studied in JET but at the applied values no appreciable effect on the runaway confinement is observed. Intrinsic to mitigation of disruptions is a means to predict that they are going to occur and also an understanding of the sequence events that ultimately leads to disruptive termination. By focussing not only on the disruptions root causes, but also on these chains, an improved probability of disruption mitigation and avoidance can be achieved. A fast valve (Disruption Mitigation Valve- DMV) has been recently installed on JET to study disruption mitigation by MGI. The results show MGI leads to significant halo force reductions and to $> 50\%$ of the disruptive energy loss being radiative. An aspect of these studies is that the MGI induced I_p-quenches tend to be faster than those occurring in natural disruptions/VDEs. Overall substantial progress has been made on JET in terms of quantifying disruption consequences and disruption control methods, leading to an improved physics basis for ITER.

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EXS/P2 — Poster Session

EXS/P2-01

Disruption Mitigation with Plasma Jets for ITER

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Disruption mitigation in ITER requires a reliable technique with real-time capability. Impurity injection has been proposed to convert the plasma energy density (~ 1 GJ in 840 m^3) into radiation power in ~ 1 ms and to increase the electron density by two orders of magnitude throughout the plasma cross section so as to achieve suppression of runaway electrons avalanche. However, once the impurity atoms are ionized in the thin outer layer of the tokamak plasma, they can no longer penetrate the confining magnetic field unless they have high velocity. Based on experimental and modeling results with injection of gas, pellets, and liquids, it seems very difficult to achieve simultaneously, deep penetration into the core plasma, efficient ablation, impurity assimilation, and an increase of electron density on the required time scale. FARTECH proposed the innovative idea of producing and using hyper-velocity (> 30 km/s), high-density ($> 10^{17}$ cm^{-3}) plasma jets of C60-fullerene. The high ram pressure of the C60/C plasma jets allows them to penetrate the tokamak hot plasma, overcoming the confining magnetic field pressure, and delivering the needed impurity mass required in less than 1 ms. For this purpose a large mass (~ 1 g) of explosively sublimated C60 molecular gas, generated by a solid state pulsed power driven source containing both TiH_2 and C60 grains, is ionized and accelerated as a plasma slug in a coaxial plasma accelerator. We present the progress in modeling and simulation of TiH_2 grains heating, C60 micro-grain powder explosive sublimation, molecular gas injection, mass separation, and plasma slug acceleration. Our 3D simulations using the LSP PIC code show that a heavy C60 plasmoid penetrates deeply as a compact structure through a transverse magnetic barrier, demonstrating self-polarization and magnetic field expulsion effects. A prototype of a coaxial plasma gun with a TiH_2/C60 pulsed power injector is under development for a small scale, proof-of-principle experiment which could be demonstrated a tokamak such as DIII-D.

EXS/P2-02

Novel Rapid Shutdown Strategies for Runaway Electron Suppression in DIII-D

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New rapid shutdown strategies have been recently tested in the DIII-D tokamak to mitigate the runaway electrons (RE). Disruptions in ITER are predicted to generate multi-MeV REs that could damage the machine. Thus the mitigation of REs is critical for the reliability of ITER. These tests included new methods to mitigate the REs generation processes, methods to allow a harmless deconfinement of the REs and new diagnostics to characterize REs. The RE generation process in ITER is expected to be mainly an avalanche process which can be mitigated at high-density levels n_{crit}. To reach this density, new particle injection schemes have been developed on DIII-D. The multi-valve massive gas injection (MGI) consists in injecting impurity gas through several fast valves. It allowed allowed reaching $0.15n_{crit}$. The shattered pellet injection (SPI) consists in injecting a large cryogenic pellet (15 mm$\times$20 mm) in the plasma shattered into small fragments by impacting on two metal plates. SPI allowed a deep penetration of the particles into the core during the thermal quench (TQ). It achieved record levels of local electron density (up to 9×10^{21} m^{-3}) during the TQ. The shell pellet injection (SHPI) which consists in injecting a polystyrene shell containing an impurity payload (boron powder or argon gas). It was tested using 2 mm and 10 mm shells. The experiments showed that the 2 mm shells burn up and release their payload deep in the core in agreement with ablation calculation, but tests using 1cm diameter shells showed that the ablation rate in this case is lower than calculated. The plasma did not burn up the shell as expected. Other strategies were also developed to harmlessly deconfine REs. The main method tested was applying a magnetic perturbation in order to disturb the RE confinement. When the magnetic perturbation was applied, the RE channel generated during these rapid shutdowns was reduced. The studies carried out to mitigate REs were completed with new diagnostics on DIII-D to characterize the RE channel. A BGO scintillators array and fast visible synchrotron imaging were implemented. These diagnostics showed that the REs have an average energy in the $10 - 20$ MeV range. They showed also that the position of the RE-wall interaction changes significantly during the whole rapid shutdown process.

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EXS/P2-03

Determination of Q Profiles in Jet by Consistency of Motional Stark Effect and MHD Mode Localization

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The performance of Tokamak advanced scenarios depends heavily on the shape of the q profile. The determination of q is only possible by combining the results from many diagnostics, which contribute to the definition of the magnetic equilibrium. The EFIT code is normally used in JET to calculate the equilibrium using data from the electrical probes and constraints from other diagnostics such as MSE, polarimetry etc.

Independent information on q can be obtained by identification and localization of magnetic islands, which can be associated to surfaces where q assumes known rational values. The radius of the island can be obtained by comparing the frequency measured by fast magnetic coils with the ion diamagnetic frequency in the laboratory reference frame. The last is inferred from the Carbon temperature and rotation measured by Charge Exchange spectroscopy. An alternative method for measuring the island position is based on the assumption that the island position corresponds to a 180 degree phase jump in the temperature oscillation measured by fast ECE radiometers.

This work is devoted to examine the consistency of the results from EFIT and from the study of magnetic islands. The main database includes shots in Hybrid scenario, an extension of the analysis to ITB discharges is under consideration. It is found that the correction to MHD data due to the velocity of the magnetic island is fundamental. The effect of the poloidal velocity is more generally considered.

EXS/P2-04

Survey into the Occurence of Disruptions and their Root Causes at JET

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Disruptions have been able to cause considerable damage in larger devices like JET, hence prevention or mitigation is essential. In order to obtain a more precise insight in the occurrence of disruptions in large Tokamaks, a detailed survey has been carried out at JET. Firstly, the average fraction of plasmas that disrupt, or disruption rate, and the so-called disruptivity, i.e. the likelihood of a disruption depending on operational parameters, have been determined. Secondly a survey into the causes of all disruptions at JET was devised. The aim of these studies is to find out what factors determine the occurrence of disruptions and what sets the lower limit. Moreover, the knowledge of the disruption causes can be used to devise better strategies to prevent and mitigate disruptions at JET and possibly in ITER.

The average disruption rates for various periods during 25 years of JET operations was found to decrease steadily and recent campaigns show an averaged rate of less than 4% while from 1991 to 1995 this was often higher than $20-30\%$. Higher disruption rates were found during exploratory campaigns while disruptions occurred much less frequently when disruption avoidance was essential, such during D-T operations. This suggests that occurrence of disruptions may partly be connected to less careful operations and human errors and the downward trend could be interpreted as a learning curve of Tokamak operations.

To shed more light on this a detailed study into the causes of all disruptions over the period from 2000 to 2007 was conducted. This allowed the visualization of the complex chain-of-events that could lead to disruptions. As expected all disruptions at JET were eventually pushed close an operational limit resulting in the on-set of physics instabilities. But it was also found that the chain-of-events were often triggered (root cause) by rather technical problems such as the failure of sub-systems such as auxiliary heating. Human error was the second highest root cause of disruptions at JET. This paper will discuss the complete set of root causes and disruption classes that were found at JET. This study provided new insights in the processes that cause disruptions at JET and provided useful suggestions on how to prevent them.

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EXS/P2-05

A Diffusive Model for Halo Width Growth During VDEs

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The electromagnetic loads produced by halo currents during vertical disruption events (VDEs) impose stringent requirements on the strength of ITER in-vessel components. A predictive understanding of halo current evolution is essential for ensuring the robust design of those components. That evolution is primarily governed by two quantities: the halo region width and resistivity. A diffusive model of halo width growth during VDEs has been developed that provides one part of a physics basis for predictive halo current simulations. The diffusive model was motivated by DIII-D observations that VDEs with cold post-thermal quench plasma and a current decay time much faster than the vertical motion (Type I VDE) possess much wider halo region widths than warmer plasma VDEs where the current decay is much slower than the vertical motion (Type II). A 2D finite element code is used to model current diffusion during selected Type I and Type II DIII-D VDEs. The model assumes a core plasma region within the LCFS diffusing current into a halo plasma filling the vessel outside the LCFS. LCFS motion and plasma temperature are prescribed from experimental observations. The halo width evolution produced by this model compares favorably with the experimental measurements of Type I and Type II halo width evolution. A closed-form, 1D model of diffusive halo width growth, which can be more simply integrated into complex simulation codes, is also presented.

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EXS/P2-06

Optimization of the Safety Factor Profile for High Non-inductive Current Fraction Discharges in DIII-D

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The safety factor (q) profile is a key parameter of tokamak discharge with 100% noninductively driven current ($f_{\mathrm{NI}} = 1$) and large fraction of self-generated bootstrap current (f_{BS}). This discharge is envisioned for the ITER steady-state mission and a DEMO reactor. The bootstrap current density (J_{BS}) is proportional to q and temperature and density gradients, while the q profile is strongly dependent on J_{BS} when f_{BS} is large. The gradients depend on transport coefficients, which depend on q and magnetic shear. Pressure limit depends on q profile and gradients. Due to complexity, it is not straightforward to identify

the optimum self-consistent q profile for $f_{\mathrm{NI}} = 1$ operation. In DIII-D the self-consistent response to changes in q profile was studied in high f_{NI}, high β_{N} discharges through a scan of q_{min} and q_{95} at 2 values of β_{N}. As expected both f_{BS} and f_{NI} increased with q_{95}. Temperature and density profiles broadened as q_{min} or β_{N} increased. A consequence is f_{BS} does not increase as q_{min} is raised above 1.5. Profile shape changes with β_{N} modify the J_{BS} profile from peaked at $\beta_{\mathrm{N}} = 2.8$ to relatively uniform in the discharge core with weak dependence on q_{min} and q_{95} at β_{N} within $10 - 20\%$ of calculated stability limit.

With J_{BS} profile relatively uniform in the discharge core and total current density (J) peaked on the axis, matching noninductive current density (J_{NI}) to the desired total J profile requires significant external current drive in the area inside the H-mode pedestal. This was provided by neutral beam current drive in these experiments. The shapes of (J_{NI}) and total J profiles are most similar at $q_{\mathrm{min}} = 1.35 - 1.65$, $q_{95} = 6.8$ where $f_{\mathrm{NI}} = 0.85$. At $q_{\mathrm{min}} = 1$ the difficulty is a significant mismatch in the area nearest the axis where $J \gg J_{\mathrm{BS}} + J_{\mathrm{NBCD}}$.

At lower q_{95} where $J_{\mathrm{NI}} < 0.7$, additional external current drive located off-axis is required to achieve $J_{\mathrm{NI}} = 1$. This current drive can be provided by off-axis beam injection planned for DIII-D in 2011, which can also enable the study of discharges with $q_{\mathrm{min}} > 2$.

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EXS/P2-07

Nonsolenoidal Startup and Plasma Stability at Near-Unity Aspect Ratio in the Pegasus Toroidal Experiment

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The Pegasus experiment is an ultralow aspect ratio spherical tokamak. The research program on this experiment is developing non-solenoidal startup and growth techniques for tokamaks, and exploring plasma stability at near-unity aspect ratio. Helicity injection from localized current sources in the plasma periphery have produced total tokamak plasma current up to 0.17 MA with less than 4kA injected. These results are consistent with a simple theory invoking helicity balance and Taylor relaxation constraints. Startup discharges created with helicity injection and poloidal field induction produce reasonable target plasmas for further current drive. For example, they readily couple to ohmic induction after helicity injection. Increasing the nonsolenoidal startup current to 0.3 MA will test theory to the point where parallel conduction losses may dominate the helicity loss rate. This regime must be addressed for extrapolation to larger fusion-scale experiments. Nonsolenoidal plasma growth following startup may be pursued via Higher Harmonic Fast Wave heating and/or Electron Bernstein Wave heating and current drive. The ability to strongly modify the plasma current profile through helicity injection and/or detailed field programming is opening a path to the unique high normalized current, high-beta regime at near-unity aspect ratio. Earlier experiments indicated a soft limit wherein the total plasma current was limited to approximately the total toroidal field rod current. Current profile

manipulation mitigates the large-scale internal tearing modes that previously limited the plasma current. This opens access to the high beta regime where the plasma current can substantially exceed the toroidal field current. Finally, operation at near-unity aspect ratio provides easy access to regimes of high peeling and ballooning mode drive in the plasma edge region. Electromagnetic filamentary structures are observed in the Pegasus edge region and they display characteristics of peeling modes. Internal magnetic probes are being used to determine the local current density profile in the plasma edge region, and thereby provide a strong test of peeling-ballooning models.

EXS/P2-08

Performance of Discharges with High Elongation and Beta in NSTX and Near-Term Paths toward Steady State

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Steady state spherical torus configurations for plasma material studies, a nuclear component testing facility (CTF), or fusion power generation, will need strong shaping, high-beta, excellent confinement and (possibly) external current drive. Recent experiments in the National Spherical Torus Experiment (NSTX) have demonstrated high-beta, high-elongation operation over a range of normalized currents in support of these needs. High poloidal-beta scenarios with non-inductive fractions of $65 - 70\%$, elongations of 2.7, normalized betas of 5, and normalized currents of 2.4, have been reliably demonstrate with minimal MHD activity, and have NSTX record low loop voltages (~ 130 mV). The density ramps to a Greenwald fraction of ~ 1, such that the neutral beam current fraction drops from $\sim 30\%$ early in the discharge to $< 15\%$ at later times; the bootstrap fraction ramps up as the density increases, achieving pressure-driven current fractions of $55 - 60\%$. These scenarios were then extended to normalized currents up to 4.2 and toroidal betas up to 30%. All of these scenarios benefited from lithium conditioning of the plasma facing components, and achieved confinement comparable to or better than ITER H-mode scaling. $n = 3$ non-resonant error field correction, $n = 1$ mode feedback were critical in maintaining stability and discharge-reliability.

Predictive modeling with TRANSP has been used to find near term means to further increase the non-inductive current fraction, starting from the high poloidal-beta discharges discussed above. A 25% decrease in the plasma density, as predicted to be possible with the recently installed liquid lithium divertor (LLD), coupled to a 18% increase in the temperature, increases the neutral beam current at the expense of the bootstrap current, and leaves the total non-inductive fraction unchanged. If the temperatures vary inversely with the density, an increase in the non-inductive fraction to 83% is predicted for this density decrease. If the densities are fixed, a 40% increase in the temperature is sufficient to achieve non-inductive operation; reducing the effective charge from three to two reduced the required temperature increase to 25%. This increase in temperature might be provided by increased confinement with LLD, or via fast wave heating in addition to that from neutral beams.

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EXS/P2-09

Approaches on Vertical Stability and Shape Control of KSTAR Plasmas in the Presence of Intrinsic Ferromagnetic Material

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In KSTAR, the choice of Incoloy908 as a jacket material of the superconductors was expected to introduce large uncertainties to the measurement of the magnetic field profiles and hence the quality of plasma controls. However, from the experiments in the 2009 plasma campaign, the effect of additional ferrite fields was strongest at the burn-through phase, but gradually decreased as the PF coil currents saturated to its higher value. It has been also shown that feedback controls of plasma current (I_p) and major radius (R_p) are not sensitive to that additional magnetic field. In order to increase controllability, the magnet system and in-vessel structures of the KSTAR were recently upgraded for the 2010 fall campaign. At first, breaking four up/down symmetric connections of the 7 pairs of PF coil system guaranteed intrinsic vertical position controls. Second, by allowing higher PF coil current and upgrading grid power, the saturation of the incoloy908 is expected to occur earlier in the discharge and its influence becomes smaller consequently. Finally, the installations of Cu in-vessel vertical stabilization coils located behind newly introduced Cu passive stabilizers increases the stability margin for magnetic controls and allow development of diverted plasmas. The design of the up/down symmetric passive plates was analyzed by dynamic simulations based on a rigid response model to determine trade-off of the $n = 0$ passive stabilizing effects for the worst case. With these resources, the initial approach to create a diverted/double-null shaped plasma is described in this paper. An experimental approach of dealing with the error field at the X-point is suggested with incorporation with the standard RTEFIT/isoflux algorithms. The final plasma pulse scenario is expected to contain a feedforward breakdown scenario provided by offline calculations, I_p ramp up plus kappa feedback until I_p = 500 kA and applications of isoflux with the X-point feedbacks. The approach is expected to be validated by separate simulations based on the 2009 data and to be illustrated with experimental results from the 2010 fall campaign.

EXS/P2-10

Real-Time Control of MHD Activity and Steady-State Current Profile by Non-Inductive Current Drive in Tore Supra

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The Tore Supra tokamak long pulse capability (superconducting toroidal magnets, actively cooled plasma facing components, large non-inductive current drive capability) has allowed a quite unique type of experiment where successive stationary states of the plasma current profile are controlled in real time during the main heating phase. The plasma current profile is controlled by varying the level of Lower Hybrid (LH) power, i.e. replacing part of the ohmic current by a non-inductive source with a different deposition profile. With the LH power available during the experiments, the safety factor profile can thus be varied at will from a sawtoothing monotonic one to a mildly reversed profile with $q_{min} \sim 3/2$. In this range, the q-profile evolves through five distinct states, characterised by specific MHD activity. The q-profiles states are detected by real time analysis of the electron temperature relaxations resulting from the MHD activity, observed on the central channels of the Electron Cyclotron Emission diagnostic (ECE). Several experiments have been carried out on Tore Supra to demonstrate the capability of this control scheme i) to obtain a desired stationary q-profile state and ii) to sustain it in spite of preset variations of other plasma parameters such as plasma density, ICRH power or total current. We achieved i) control of the presence/absence of sawteeth with varying plasma parameters, ii) obtaining and sustaining a “hot core” plasma regime without MHD activity, iii) recovery from a voluntarily triggered deleterious MHD regime.

We report also on another set of experiments featuring real-time control of Electron Cyclotron antenna mirrors for destabilisation of sawteeth in the presence of fast ion tails produced by ICRH, using an improved ECCD positioning algorithm. This experiment shows that achieving and maintaining short sawteeth through real time controlled ECCD seems a viable option for NTM avoidance in plasmas where sawteeth are stabilised by significant fast ion pressure inside the $q = 1$ surface.

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EXS/P2-11

DIII-D Experimental Simulation of ITER Scenario Access and Termination

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Reliable access to, and termination of, burning plasma scenarios is crucial for the success of the ITER mission. DIII-D has simulated both the startup and rampdown phases by scaling the ITER shape and plasma current penetration time to DIII-D experimental conditions, and has improved scenarios. In the startup phase, 2nd harmonic EC assist has allowed prompt and reproducible burnthrough and has been compared to the more conventional startup with Ohmic heating alone. Both low field side and high field side limited discharges have been compared during the current ramp phase and scans of initial vertical field, EC resonance location, EC launch angle, neutral pressure, and atomic species (D_2 and helium) have been carried out to evaluate the potential operating space for ITER. Ramp-down to DIII-D scaled values below the ITER prescribed limit of 1.4 MA have been successfully achieved, both from the ITER scenario 2 flattop phase and also the 17 MA scenario. This ramp-down was demonstrated while holding the strike points approximately fixed, corresponding to the divertor region in ITER, and reducing elongation. Algorithms for vertical feedback have been tested to successfully achieve this low I_p "soft landing". A faster rampdown than the original ITER scenario was shown not to consume additional flux, allowing ITER more flexibility to attain its specified current flattop duration. A 20% reduction in rampup flux was also demonstrated with modest additions of auxiliary heating power. The Corsica-free boundary equilibrium code has been used to simulate these DIII-D discharges. Corsica modeling of these discharges indicates that the internal inductance is most sensitive to the conductivity in the outer portion of the discharge.

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EXS/P2-12

Stable Plasma Start-up in the KSTAR under Various Discharge Conditions

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The goal of tokamak start-up treated in this research is to raise the plasma current up to 100 kA in the KSTAR device. Start-up of the KSTAR device differs from typical tokamak start-up in two aspects. First of all, the KSTAR device requires a Blip Resistor Insertion System (BRIS) to produce a sufficient change of poloidal field (PF) coil current due to the limited voltage capability of the power supplies. The other unconventional aspect of KSTAR start-up originates from the ferromagnetic material incoloy 908 used in the PF and toroidal field (TF) coils. In order to use the BRIS, the KSTAR start-up scenario utilizes a delicate counter-balance among the outer PF coils. Therefore the ferromagnetic effect becomes more significant under the circumstance of subtle field balance during plasma start-up. In detail, the radial gradient of the vertical magnetic field is strongly deformed due to a relatively strong ferromagnetic effect near the inboard side. As a result, most of the failed shots in the KSTAR 2009 experiments exhibited an inboard crash during the initial stage. It seems that the start-up scenarios used in 2009 could not guarantee the stability of the radial position because the scenarios do not include the ferromagnetic effect. As a matter of fact, the establishment of tokamak start-up scenarios necessitates time-dependent calculations with consideration of eddy currents in conducting structures such as the vacuum vessel and passive stabilizers. Nonlinear ferromagnetic effects make the problem difficult because numerical methods based on the Green's function cannot be directly applied in calculating the magnetic field structure. In order to overcome this difficulty, time-dependent simulation is mimicked by a series of nonlinear ferromagnetic calculations with given PF coil currents and eddy currents at specific times. Prior to these nonlinear calculations, time-dependent eddy currents were obtained by solving coupled circuit equations based on the Green's function method. Systematic efforts are being conducted to establish robust start-up scenarios with compensation of the above mentioned ferromagnetic effects. The developed scenarios will be demonstrated in the KSTAR 2010 campaign.

EXS/P2-13

Disruption Mitigation by Massive Gas Injection in JET

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Disruption mitigation is mandatory for ITER in order to reduce forces and to mitigate heat loads during the thermal quench (TQ) and by runaway electrons. A fast disruption mitigation valve (DMV) has been installed at JET to study mitigation by massive gas injection (MGI). Different gas species have been investigated: D2, Ne, Ar, He and mixtures of Ne and Ar with 90% of D2. A maximum of 2.5×10^{23} particles has been injected, corresponding to about 100 times the electron content in the plasma.

Halo currents are successfully reduced by MGI, if the TQ is initiated before a significant vertical displacement has taken place. With the minimum reaction time of 6 ms for the Ar/D_2 mixture a reduction by 60% has been achieved. The heat loads are reduced by MGI through enhanced radiation. Up to 50% of the initial thermal energy are lost before the TQ by radiation. About 40% of the remaining energy are radiated during the TQ. Only 30% of the initial energy are lost by convection, most of which to main chamber components. The radiation is homogenously distributed poloidally, with a peaking factor below 1.5 during thermal and current quench. Significant poloidal peaking of up to 2.5 is observed only before the thermal quench. Be melting at the injection port during this phase could be an issue for ITER and will be addressed in this paper. Runaway generation is successfully avoided by the injection of D_2 mixtures, due to the suppression of the Dreicer mechanism. In contrast, injection of pure Ar leads to runaway generation even at low toroidal magnetic fields, caused by a low fuelling efficiency of heavy gases. Although, runaways can be safely avoided by MGI in JET disruptions, the density reached is a factor 50 below the critical density for avalanche suppression, which is essential in ITER.

Extrapolation from JET MGI experiments towards ITER requires, on one hand, a size scaling of experimental data and, on the other hand, modelling of the processes during

MGI. We compare in this contribution the JET results with MGI experiments performed at MAST and TEXTOR. This comparison is facilitated because all three devices operate the same type of DMV, for which the gas flow behaviour is known from lab experiments. The JET data will be used to validate MGI modelling by the 2d fluid code TOKES and by simple models.

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EXS/P2-14

Suppression of Runaway Electrons during Disruption in HT-7

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One of the important problems of a large tokamak such as ITER is the disruption generated runaway electrons, which impinge the plasma facing components (PFC) and damage them. The experiments for mitigating and avoiding the current quench and runaway electrons during disruption have been carried out in HT-7 by LHW, massive gas injection and magnetic perturbation.

The plasma current quenching time is typically $1 \sim 2$ ms for a major disruption in HT-7. When LHW was injected, the post-disruption current with a plateau can be sustained up to a soft termination of discharge. Current carried by LHCD driven electrons plays an important role in this operation scenario.

Above experiments shows clearly that LHCD can be used to mitigate the current quench of a disruption, but runaway electrons were still generated. Furthermore, to suppress runaway electrons, massive deuterium is injected into plasma together with injection of LHW in HT-7. The plasma density is significantly increases with gas puff during the post-disruption current plateau. The amount of runaway electrons in this scenario is reduced compared to the discharge without gas injection after major disruption.

Another way tried in HT-7 to suppress runaway electrons generated during major disruption is by magnetic oscillations. Radiation of runaway electrons nearly disappeared when strong magnetic oscillations exist. It seems to be the most effective way to suppress runaway electrons in HT-7.

The underlying physical mechanisms from these experiments is being analyzed and discussed in detail. These techniques for suppressing runaway electrons during major disruptions will be further verified in EAST, which is equipped with more diagnostics and has more capability for these investigations.

EXS/P2-15

Contribution of ASDEX Upgrade to Disruption Studies for ITER

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Plasma disruptions represent a hazard for the structural integrity of ITER, since they will generate runaway electrons, large mechanical forces and thermal loads. The contribution of the existing tokamaks to ITER consists firstly, in refining the characterization of these loads and their extrapolation, on the basis of physical models, and secondly, in learning to predict and mitigate disruptions. The ASDEX Upgrade (AUG) research program covers these specific topics and this contribution reports on significant progress made recently in these areas.

The formation and evolution of the halo region is analyzed and the disruption database is used to find out general characteristics in the spatial distribution of the halo current. This work is part of the effort made by the existing tokamaks in the framework of the ITPA MHD Topical Group to characterize the development of the halo region on the basis of experimental measurements and physical models. In particular, it supports the detailed benchmark of VDEs carried out with the MHD-transport codes DINA and TSC.

The disruptions of the last four years have been analyzed and classified accordingly to their physical causes. Discriminant analysis is applied to each disruption group in order to determine the most significant plasma variables, which allow for classification, and is being used to discern between stable and pre-disruptive plasma states.

Progress has been made in AUG in attaining an effective electron density equivalent to 24% of the critical one, necessary for the collisional suppression of runaway electrons, while the fuelling efficiency remains at the level of 20% for plasmas with a modest thermal energy. Nevertheless, a significant decrease of the fuelling efficiency with increasing plasma thermal energy has been observed at large amounts of injected helium and neon atoms. Independent of the thermal energy, the density distribution is poloidally and toroidally very asymmetric, implying that multiple valves are needed to raise the density further. The fast current decay and the slow impurity redistribution in the plasma, observed when large amounts of gas are injected, constitutes physics limits, which could prevent the applicability of this method to runaway electron suppression in ITER.

EXS/P2-16

Disruption and Runaways Electron Mitigation Studies on Tore Supra

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Disruptions and runaway electrons (RE) are identified as a major issue for ITER and reactor-size tokamaks. Disruption produces excessive heat loads on plasma facing components, induces strong electromagnetic forces in the vessel structures, and generates multi-MeV runaway electrons. Thus mitigation techniques are strongly mandatory to minimize disruption and RE effects on plasma facing components (PFC). Massive gas injection (MGI) prior to thermal quench and/or during the current quench is one of the proposed methods studied on Tore Supra. Comparisons of different gases (He, Ne, Ar, He/Ar mixture) and amounts (5 to 500 Pa·m^3) were made to investigate gas jet penetration and mixing, showing that light gases are more efficient regarding runaway electron suppression than heavier gases. Eddy currents in the limiter are moderately reduced by all the gases, and may be more dependent on the time constants of the structures than on the gas species. Gas mixing before the thermal quench is better with light gases. Gas jet penetration is observed to be shallow and independent on the gas nature and amount. The gas cold front is stopped along the $q = 2$ surface where it triggers MHD instabilities, expelling thermal energy from the plasma core.

After unmitigated disruption occurring during the current ramp-up, a long RE tail (up to several seconds, several hundred of kiloAmps) may develop on Tore Supra. This provides a unique opportunity to characterize plasma mainly sustained by relativistic electrons. Several mitigation techniques are thus under evaluation to reduce PFC loads:

- control of the RE beam position such as driving the RE on dedicated PFC,
- time spreading of the energy loss associated to a decelerating electric field,
- use of MGI (He and Ar) for slowing-down the RE are investigated.

An active position control of the RE current barycentre has already been successfully tested. First results show that electric deceleration is not very effective and request a better control of the RE current. MGI seems more promising. The photoneutron production, used as an indicator of the amount of RE hitting the wall, is increased after MGI without wall damages. The RE transverse transport is increased and a part of photoneutrons is also generated in the middle of the vacuum chamber, both minimizing the final amount of energy density deposit on the wall.

EXS/P2-17

Effects of ECH/ECCD on Tearing Modes in TCV and Link to Rotation Profile

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This paper will focus on the MHD modes observed on the TCV tokamak, their sensitivity to local ECH and ECCD deposition and their role with respect to the measured toroidal rotation profile. The first part is directly related to the classical tearing mode parameter D', which is a key value for predicting NTMs in ITER. In TCV L-mode plasmas, tearing modes are usually not unstable but they can be destabilized with localized co-ECCD at very specific radial positions. It is shown that counter-CD at the same position is stabilizing. These plasmas are used to determine the sensitivity of the tearing mode triggering to the current profile modifications. This is then compared to the expected q profile variation required to stabilize these modes, as obtained from NTM modeling. Calculations using PEST-III and a cylindrical calculation of D', using the q profile, have been performed. The observed sensitivity is indeed very high and small changes can lead to significant positive values of D', and thus destabilize the tearing modes. These simulations have shown that the value of q, its 1st and 2nd derivatives but also the position of the ECCD perturbation with respect to the rational surface all have a significant influence on D'.

The role of ECH and/or of MHD modes on the toroidal rotation profiles are analyzed, in particular the role of MHD on the rotation inversion observed in TCV L-modes. The rotation inversion is always observed when q_{95} is close to 3 and it is known that these discharges are more prone to MHD. In many cases the inversion occurs with the onset of a stationary tearing mode. The inversion is also observed at lower plasma current and density when the plasma is forced from a limited to a diverted configuration. It is found that during the transition a significant modification of the current profile occurs which can lead to MHD activity. Several scenario optimization tools - modifying the transition from a limiter to a diverted plasma such as to avoid the onset of tearing modes, or the addition of ECH to stabilize or destabilize a mode - are used to decouple the effects of high current and of MHD modes on the stationary toroidal rotation profiles. Several effects occur simultaneously and detailed experiments are required to study the link between MHD and rotation.

EXS/P2-18

Optimization of EAST Plasma Start-up for Simulations of ITER with Low Loop Voltage

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The Experimental Advanced Superconducting Tokamak (EAST) was the first shaped tokamak of mega-ampere plasma scale utilizing a fully superconducting poloidal and toroidal field coil system. Owing to its similarities in coil configuration and constraints to ITER, it can readily determine issues of breakdown, plasma formation and shaping, and plasma current ramp specifically relevant to ITER. Plasma start-Up have been optimized to minimize superconducting PF coil AC lose for simulating ITER safe operation in future. EAST achieved reliable Ohmic plasma breakdown and initial ramp-up at toroidal electric field of 0.5 V/m. In 2009, we successfully realized the plasma initiation assisted by lower hybrid wave under toroidal electric field ~ 0.16 V/m. Plasma ramping rate between 0.1 MA/s and 0.5 MA/s have been obtained with different shaping and assistance of LHW. We describe both routine plasma startup scenarios and plasma startup in low electric field aided by LHW in EAST. Control design tools and plasma control algorithms utilized during the first campaign are briefly discussed. Key physics, engineering, and operation experiences including observations relevant to future devices such as ITER will be presented.

EXS/P2-19

Double Null Merging Start-up Experiments in the University of Tokyo Spherical Tokamak

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University of Tokyo Spherical Tokamak (UTST) is a unique device in the world that demonstrates merging start-up of the spherical tokamak (ST) plasma by using the poloidal field (PF) coils located outside the vacuum vessel. When two ST plasmas merge together to form a single ST, their magnetic field lines reconnect, converting the magnetic field energy to the plasma kinetic or thermal energy and increasing the plasma beta up to 50% within the short reconnection period. The ST merging experiments have been demonstrated in the START/MAST (UKAEA) and TS-3/4 (Univ. Tokyo) devices, by using the PF coils located inside the vacuum vessels. Here, we present the successful generation of

the spherical tokamak started-up using the external PF coils outside the vacuum vessel. This double null merging (DNM) method has been successfully performed in UTST. Two initial STs with aspect ratio of 2.4 were formed at the null-points and were merged to form a single ST with aspect ratio of 1.5. The plasma current of 40 kA and the pulse width of 0.6 ms were obtained. After the completion of the plasma merging, the thermal energy reached 80 J (central pressure of 200 Pa), and the heating power of reconnection was as high as 1.7 MW. In UTST, a central solenoid (CS) coil (95 turns, 45 kJ) is installed to improve the plasma parameters for incoming neutral beam injection and radio-frequency heating. In the case that using CS in addition to the DNM discharge, the plasma current and discharge time increased to 90 kA and 1.2 ms, respectively. The thermal energy reached 230 J (central pressure of 400 Pa) after the completion of the merging, and the heating power was as high as 4.5 MW. Note that the effect of CS is included in this power. After the merging completion, the thermal energy continued to increase and reached 650 J (central pressure of 1000 Pa). The DNM method is more reactor-relevant for developing the future fusion reactor. The UTST experiment has developed a useful CS-less start-up scheme, which is world-widely studied in the spherical tokamak research.

EXS/P2-20

First Observation of Persistent Small Magnetic Islands on HL-2A

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A spontaneous and persistent small island has been observed in the most of HL-2A discharges, which challenge the existent theoretical model for the threshold for NTMs' onset. The small MHD activity can exist for the whole discharge period and keep constant amplitude and frequency under the unchanged plasma parameters. A large $m/n = 2/1$ island (NTM), growing from a small 2/1 mode coupled with an $m/n = 1/1$ oscillation in ECRH heated discharge has been found. For understanding the experimental results, numerical modelling has been carried out by using nonlinear two-fluid equations and taking into account the bootstrap, ion polarization current and the perpendicular heat transport. With experimental data as input, two nonlinear regimes are found: a small island regime existing for little or no bootstrap current and a large one (NTM regime) existing for a sufficiently high bootstrap current fraction. The numerical results agree with experimental observations, providing new insight into the mechanism for the NTM's onset. The significance of the observation is that the small magnetic island is essentially existent in tokamak discharges. This is different from the description of previous theory. The small island can be a seed island and develop to an NTM when the island width exceeds a threshold by the coupling with other events, e.g. $m/n = 1/1$ mode, sawteeth and ELM, etc.

EXS/P2-21

Experimental Study of Electron Scale Density Fluctuation in LHCD Plasma on HT-7 Tokamak

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The behavior of the electron anomalous heat transport is a critical issue the reversed shear configurations in ignited plasma, where a-particle heating raises the electron temperature above the ion temperature. On the HT-7 tokamak, a CO_2 collective scattering system was installed to investigate electron scale micro turbulence. Three independent channels with different scattering angles are used simultaneously to monitor density fluctuation. The system can measure density fluctuation with $k_\theta \rho_s = 1-4$ or TEM (trapped electron mode) and TEM/ETG (electron temperature gradient)scale. The power spectra were measured as a function of wave-numbers in different plasma parameters. The medium- and high-k domains density fluctuations were observed with increasing low hybrid power. It was found that the stability of modes in TEM range and ETG range should be related with the plasma current profile, and the normalized frequency spectrums were independent with the low hybrid power before the e-ITB formed. However, a power threshold may exist. When the low hybrid power exceeded the threshold, the shape of frequency spectrum was not changed in channel $k_\theta = 12$ cm^{-1}, but the shape of frequency spectrum in channel $k_\theta = 24$ cm^{-1} was obviously changed when low hybrid power exceeded 500 kW. The transition into the PLH> 500 kW phase seemed to be attributable to the TEMs stabilization by the reversed current profile formed by LHCD low hybrid current drive, whereas the steep ETG destabilized the small scale ETG modes. Abel inverted hard X-ray emission profiles of shot #106196 could be used to reconstruct the total current profile in this non-inductive state sustained with low hybrid power because residual ohmic current was small. And the electron temperature profiles given by Thomson scattering system developed a steep gradient near $r/a \sim 0.4$. However, beyond the power threshold the fluctuation frequency spectrum of two channels changed and an e-ITB may be produced. In another series shots of increasing low hybrid power, the eruption in micro-instability appeared at a lower hybrid power larger than 475 kW with a plasma density smaller than 10^{19} m^{-3} was observed. When low hybrid power exceeded the threshold, the density fluctuation level enhanced and the frequency spectrum was broadened.

EXS/P2-22

Study of Edge Turbulence from the Open to Closed Magnetic Field Configuration during the Current Ramp-up Phase in QUEST

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Statistical feature of perturbations and blob propagation have been studied using the high speed visible light images during the RF plasma production and current ramp-up phase. In a mirror magnetic configuration with vertical (B_z) and toroidal fields (B_t) helix-sinusoidal perturbations are exited, whose helix angle and vertical wavelength are consistent with those of the magnetic field lines there. Statistical analysis shows a quadratic relation between kurtosis and the skewness over the wide region and condition. Near the scrape off layer boundary of the ohmic plasma helical perturbations are also excited. It is found that intermittency becomes significant depending on the distance from the last closed flux surface. The radial velocity of the blob is ~ 1 km/s in the slab plasma, and poloidal propagation velocity is ~ 1.5 km/s in the SOL. Radial acceleration of a blob is found in the slab plasma.

EXS/P3 — Poster Session

EXS/P3-01

Controlled Resonant Magnetic Perturbation Physics Studies on EXTRAP T2R

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Resonant magnetic perturbations (RMP) are now an essential tool in fusion plasma physics to address severe problems related to edge localized modes (ELMs), magnetic islands and neoclassical tearing modes (NTMs). While recent experimental work focuses mainly on the RMP effect on global plasma parameters, from an experimental point of view, the effect of an RMP on the TM dynamics and on the plasma flow is not yet studied in a controlled fashion. The present work describes the contribution of the EXTRAP T2R reversed-field pinch (RFP) to the study of the RMP effect on the TM dynamics and on the plasma flow.

The work first describes the algorithm used produce external magnetic perturbations with modern feedback tools. This new feedback algorithm, called revised intelligent shell (RIS) exploits control theory methods to apply external perturbations with desired characteristics. Basically, the intelligent shell concept is treated in a generalized way that enables “output-tracking”. The user can decide a reference signal for each harmonic individually (time-dependent amplitude and phase) and the feedback will control the magnetic boundary to produce the desired harmonic and thus reproduce the reference signal at the sensor coils.

The paper then focuses on the physical results. The RIS algorithm is used to apply a stationary magnetic perturbation and to study its effect on a TM rotating with an angular velocity of 40 krad/s. Also the resulting slowing-down effect on the plasma flow is measured. The RMP can induce a modulation in the TM amplitude time evolution and in the TM rotation velocity. If the RMP amplitude is sufficiently large, then phase jumps occur. Effects on the plasma flow are studied by monitoring the velocity of impurities (such as OV, mainly concentrated in the EXTRAP T2R plasma core). It is shown that during the RMP phase the plasma flow is braked and that soon after the RMP is removed the flow recovers its natural value. Braking is also observed if a non-resonant external perturbation is applied. Although the perturbation is non resonant, clear evidence of plasma flow braking is measured. The use of the RIS algorithm thus enables controlled studies of neo-classical toroidal viscosity.

EXS/P3-02

Study of Edge Localized Mode in HL-2A Tokamak Experiments

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In HL-2A divertor configuration, the ELMy H-mode operation was first achieved in 2009 experiment campaign, by combining the auxiliary heating of NBI (< 0.8 MW) and ECRH (< 1.2 MW) with 2nd harmonic X-mode, which are strongly dependent on the optimized discharge conditions. The ELMs are detected by distinctive spikes in the divertor Da radiation. It is obviously that ELM events exhibit irregular behavior. After L-H transition, the central-chord-averaged electron density increase and the density peaking factor decrease, thus the pedestal density may increase. The core plasma electron temperature almost keeps unchanged, so the pedestal electron temperature may also keep unchanged. The useful method to categorize ELMs by comparing the pedestal temperatures and densities of plasma discharges is utilized. The ELM occurs at high pedestal density and low pedestal temperature, in a similar region to where the L-H transition occurs. As a result, the ELM of present HL-2A H-mode operation is recognized as Type III, its typical frequency is about $400 \sim 600$ Hz. As a first H-mode operation on HL-2A, the available data are not enough to confirm the conclusion. Further study on the dependence of ELM frequency on heating power is in progress in present experiment campaign. The preliminary results show that Type I and Type III ELMs were observed in the experiment. Unstable periodic orbits (UPOs) in the ELM time series have been observed, showing that a deterministic, chaotic process governs the apparently random distribution of the delay between ELMs in HL-2A tokamak. The ELM plasma released from the separatrix is evolved into filamentary structures in SOL region. During a single ELM, the ion saturation current (ISAT) of the mid-plane Langmuir probe for blob studies, as well as the signal of Mirnov coil, has many bursts. Each individual burst is considered as a single filament. Radial and poloidal velocity of filaments are calculated from time delay between the maximum of the ISAT measured by the radially and toroidally separated Langmuir pins. The spatial-temporal properties of filament densities in SOL region are compared with those inter-ELMs. The time interval between ELMs and the duration of an ELM are very similar to those of Type-III ELMs in the ASDEX tokamk.

EXS/P3-03

ELM Pacing Investigations at JET with the New Pellet Launcher

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Pellet pacing is one approach under investigations aiming to reduce the energy per ELM by raising in a controlled manner the ELM frequency. This should be achieved with the smallest impact on all other plasma parameters besides ELM frequency. A new pellet system was developed for JET, designed to demonstrate pellet pacing under plasma conditions nearest to ITER with the lowest fuelling impact. Although not yet reaching the designated performance, the system was able to further validate the pellet pacing approach to ELM size control. A pacing approach in the ITER baseline scenario confirmed a tenfold ELM frequency enhancement will be possible once the pellet system reaches its designated parameters. In a low triangularity configuration pellets gain control with $f_{Pel} = 10$ Hz $= f^0_{ELM}$ but ELMs fully synchronized to pellet injection. All sound pellets do trigger ELMs, almost all ELMs are triggered. Hence, ELM pacing was established confirming the technique in the large size tokamak JET. Investigations of the dynamics of triggered ELMs and their spontaneous counterparts but also of the pure pellet plasmoid perturbation in cases no ELM was triggered were performed. Observations are in agreement with a proposed triggering mechanism assuming the first filament releasing the ELM grows directly from the high pressure plasmoid generated by the pellet. It is thought the minimum possible pellet size and hence fuelling constraint is evoking from the minimum pellet penetration required to establish a sufficient perturbation for the triggering. Using the pacing section of the launcher for the first time trigger investigation in the according pellet size regime became possible. A dedicated experiment with inboard pellet launch showed a clear mass trigger threshold around $0.1 - 0.2 \times 10^{20}$ D. These small pellets are not found to contribute any net plasma density increase. Results indicate a pellet must proceed sufficiently deep (several cm) into the pedestal to trigger an ELM, shallow (few mm) penetration is insufficient. This finding is also confirmed by recent investigations from AUG. A more detailed analysis of the experimental trigger conditions, taking also into account initial plasma conditions, is under way and will be presented as well as a discussion of the possible consequences for ITER.

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EXS/P3-04

Multi-Resonance Effect in Type-I ELM Control with Low n Fields on JET

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Multiple resonances in the Edge Localized Mode (ELM) frequency (f_{ELM}) as a function of the edge safety factor q_{95} have been observed for the first time with an applied low $n = 1$, 2 field on the JET tokamak. Without an $n = 1$ field applied, f_{ELM} increases slightly from 20 to 30 Hz by varying the q_{95} from 4 to 5 in a type-I ELMy H-mode plasma. However, with an $n = 1$ field applied, a strong increase in f_{ELM} by a factor of 4.5 has been observed with resonant q_{95} values, while the f_{ELM} increased only by a factor of 2 for non-resonant values. This result suggests that there are two effects of the RMP on the ELM frequency, one which has no q_{95} dependence, resulting in a relatively weak increase of f_{ELM}, and a second which depends strongly on q_{95} and causes a stronger increase of f_{ELM}. We may call the first one a global effect, and the second one is the so-called multi-resonance effect described in this paper. These two effects are most likely due to different physics mechanisms. A model, which assumes that the ELM width is determined by a localised relaxation triggered by an unstable ideal external peeling mode, can qualitatively predict the observed resonances when low n fields are applied.

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EXS/P3-05

Observation of Confined Current Structures in JET High Temperature Pedestals and Transient ELM Suppression

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In an effort to approach the pedestal conditions to be expected in ITER studies of high temperature pedestal ($T_{e,ped} > 2.5$ keV) plasmas were carried out at JET. In such plasmas the MHD phenomenon known at JET as an Outer Mode (OM) was observed to appear spontaneously when fuelling and recycling were low and NBI heating relatively high. While present, the OM can clamp the pedestal pressure and substantially delay ELMs, producing good confinement in a quasi-stationary state, possibly related to the QH-mode.

The longest-lived OM was observed in a Hot Ion H-mode. This OM begins after the L to H transition and delays the appearance of the first ELM up to 1 s (confinement time is of order 400 ms, both during OM and during ELMy steady phase). The OM leads to a reduction in the rate of rise of density in core and pedestal, pedestal top pressure and rotation become stationary, the power outflux is regularised. Simultaneously with OM appearance, the Hot Ion H-mode regime is lost and confinement reduces to $H_{98y} \sim 1$. After an initial transient drop Wdia stays approximately constant during OM and ELMy phases. The OM appears to be terminated by the slowly rising pedestal density, or by the reduction in edge rotation shear. OMs can also be terminated by ELMs. When well established, the OM is a stable feature of the plasma, surviving sawtooth crashes, but not large ELMs.

A detailed investigation of the OM leads us to conclude that it is a confined closed co-current ribbon, located at the flat-top of the pressure pedestal, co-rotating toroidally with the plasma.

The time-averaged heat flux to the divertor target during the OM and ELMy phases is very similar, it can be as high as $10-30$ MW/m^2. A continous OM, if it can be sustained, could become a useful operating regime for ELM elimination.

EXS/P3-06

Non-resonant Magnetic Braking on JET and TEXTOR

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Non-resonant magnetic braking effect induced by the non-axisymmetric Resonant Magnetic Perturbation (RMP) fields is investigated on JET and TEXTOR. The collionality dependence of the torque induced by the $n = 1$ RMP field is obtained on JET. The measured torque is mainly in the plasma core regime and it is about half of the NBI torque. The calculation shows that the ions are mainly in the $\nu - \sqrt{\nu}$ regime and the electrons are mainly in the $1/\nu$ regime. The observed rotation damping rate scales as $\sim \nu_i^{-0.64}$, which is contradict with the NTV theory in the $\nu - \sqrt{\nu}$ regime. The magnitudes of the NTV torque is still at least one order smaller than the observed ones on JET, although the Lagrange variant of the magnetic field strength is used and the electron NTV is included. In the plasma core region on JET, the drift frequency of the curvature and gradient of magnetic field is close to the electric drift frequency. The superbanana effects may further enhance the NTV torque, which is to be discussed. There is no obvious braking effect with $m/n = 6/2$ DED on TEXTOR, which is consistent with NTV theory. The non-resonant magnetic braking effect strongly depends on the coils configuration. To avoid it, the RMP coils should reduce the non-resonant harmonics and be located as close as possible to the plasma or use a high n number so that the field rapidly decays inside the plasma.

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EXS/P4 — Poster Session

EXS/P4-01

Momentum Transport In TCV Across Sawteeth Events

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The scaling of toroidal rotation has been studied over a wide range of in plasma and shape parameters on the TCV tokamak using a diagnostic neutral beam that injects a negligible momentum into the plasma. As on other Tokamaks, the toroidal rotation magnitude on TCV, for a given shape and plasma conditions, is observed to scale inversely with the plasma current. More specifically, this describes the maximum toroidal rotation in plasmas with strong sawteeth activity. The toroidal rotation gradient from the plasma edge was similar over a wide range of plasma currents and only diverged inside the sawtooth inversion radius. By extrapolating the maximum core rotation from the rotation profiles outside the sawtooth inversion radii and adding a correction for the observed ion temperature changes, the scaling of the intrinsic toroidal rotation with plasma current is no longer observed.

The TCV CXRS diagnostic was configured to measure the toroidal rotation profile with a low spatial resolution (4 points across the radial co-ordinate) and a high temporal resolution (2 ms). Using ECH X2 power deposited close to the sawteeth inversion radius, the sawteeth repetition time was extended to over 12 ms. The measurement used a newly available Real-Time node to generate a sequence of trigger pulses from analysis of the soft X-ray such that each spectroscopic frame was taken at 2, 4, 6, 8, ... ms after each sawtooth. The radial toroidal rotation profile evolution over a sawtooth was determined using coherent resampling over many sawtooth events.

Initial results indicate that, at the sawtooth crash, the plasma core accelerates in the co current direction and then the core rotation profile relaxes back to the counter current direction with a time constant ~ 10 ms. This implies that the scaling of intrinsic rotation with plasma current may be better understood as a peaked rotation profile (or an inwards momentum pinch) in the plasma core region that is flattened by sawtooth activity. This change of viewpoint should be integrated into the theory of momentum generation and transport in tokamaks such as ITER in that these mechanisms are not as universally affected by the plasma current as was at first thought.

EXS/P4-02

Plasma Potential and Turbulence Dynamics in Toroidal Devices: Survey of T-10 and TJ-II Experiments

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The direct study of the electric potential and its fluctuations has been done for comparable plasma conditions in the T-10 tokamak ($B = 1.5 - 2.5$ T, $R = 1.5$ m, $a = 0.3$ m, $P_{ECRH} < 2.0$ MW,) and TJ-II stellarator ($B = 1$ T, $\langle R\rangle = 1.5$ m, $\langle a\rangle = 0.22$ m, $P_{ECRH} < 0.6$ MW, $P_{NBI} < 0.9$ MW) by HIBP. In spite of the substantial differences in T-10 and TJ-II magnetic configurations, the electric potential fi has the following important similar features: (i) The scale of several hundred Volts; (ii) When $n_e > 1 \times 10^{19}$ m^{-3}, the potential has negative sign and comparable values in spite of the different heating methods: OH and ECRH in T-10, ECRH and/or NBI in TJ-II; E_r also has the same scale in both machines; (iii) The rise of n_e or τ_E is accompanied by changes of ϕ and E_r profiles towards the negative side. ECRH and associated T_e rise and τ_E degradation cause the changes of ϕ and E_r towards the positive side; (iv) Finally, negative ϕ and E_r characterize the regimes with better confinement. In addition, Geodesic Acoustic Modes (GAM) in T-10 and Alfven Eigenmodes (AE) in TJ-II exhibit dozens Volts potential oscillations and potential-density coherence. While GAMs modulate the high-frequency turbulence, AE contribution to the bulk plasma turbulent particle flux is small in comparison with the broadband turbulence flux.

EXS/P5 — Poster Session

EXS/P5-01

Advanced Control of MHD Instabilities in RFX-mod

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The RFX-mod device is particularly suited to explore innovative concepts in MHD control by means of active coils thanks to the power and the flexibility of its system.

Tearing mode dynamics under feedback controlled conditions proved to be highly non-linear. In the last two years, an optimization of the feedback laws have been performed with the help of the RFXlocking code, which include the non-linear dynamics of TMs, in order to keep to the lowest possible level the edge field of TMs and to reduce the transient error fields. Further improvements in TM control are obtained by refining models that include toroidal geometry and deviations from uniformity of the passive structures. First steps in this direction have been implemented, while the more elaborated dynamical pseudo-decoupler approach, which takes into account the frequency dependence of the toroidal coupling between actuators and sensors, is under development. The RFXlocking code has also been used to investigate the possibility to improve tearing mode control with different coil geometries.

Resistive Wall Mode control studies on the last two years focussed on indentifying and solving problems common to RFP and tokamak configurations. The issue of mode non-rigidity was studied, in particular testing the influence of coil geometry and number. The 192 active coils set can be on purpose downgraded and reconfigured, to test the role of active coils geometry and number on mode stabilization. Significant advances were produced also in the modelling of RWM control: a new integrated simulator for closed loop control experiments was developed and benchmarked. The tool couples self-consistently a full 3D description of the machine boundary (Cariddi code), a 2D toroidal model of stability (MARS) and a dynamic model of the control system cast in the state variable representation and proved to reproduce accurately experimental results.

Preliminar ohmic tokamak experiments at $q(a)$ below 2 have been performed without observing disruptions in feedback controlled conditions.

In conclusion, important advances on MHD active control have been accomplished during the last two years in different areas of active MHD control, fostering in addition interest and collaborations from many external laboratories.

EXS/P5-02

Kink Instabilities in High-Beta JET Advanced Scenarios

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Stability of high-beta plasmas is studied on discharges from a series of JET experiments on steady state and hybrid advanced scenarios, with q_{95} varying from 4 to 5 and central q from 1 to above 2. The explored range of normalized beta extends up to 4. The main macroscopic instabilities in these high β_N plasmas are kink modes with toroidal number $n = 1$ and tearing modes with n up to 6. The focus of this paper is on $n = 1$ modes, which have serious consequences on confinement and, in their strongest forms, can initiate major disruptions. Since mode frequencies are in the kHz range, well above the inverse wall time (about 10 ms), the instability should be of kink-infernal rather than of external kink nature. The main motivation for this study is that plasmas with $1.5 < q(0) < 2$ can be unstable at $\beta_N = 2.5$, while β_N up to 4 can be reached with lower $q(0)$. Stability boundaries in terms of $q(0)$ and pressure peaking are determined, and implications for ITER advanced scenarios are discussed, including interpretation of Resonant Field Amplification, that has been measured over a broad $q(0)$ range. The observed $n = 1$ modes feature global extent, lack of magnetic reconnection and ballooning character. Two different forms of time development are identified, one transient with chirping frequency and one continuous. Global and ballooning characteristics of the observed $n = 1$ modes are consistent with ideal kink-ballooning (infernal) instability, for which wall stabilization should be weak in spite of relatively large rotation frequency. Mode structure and instability domains are compared with MISHKA code predictions. The effect of the wall, damping due to energetic/thermal populations and diamagnetic effects are also assessed. Frequency chirp of fishbone-like bursts indicates that energetic ions could contribute to mode drive, in fact the frequency range is consistent with precession of trapped energetic ions; this contribution is evaluated by the HAGIS stability code. The role of nonlinear saturation due to field line bending is also evaluated.

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EXS/P5-03

The Impact of 3D Fields on Tearing Mode Stability of H-Modes

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New processes have been identified in the interaction of 3D fields with tearing mode stability in tokamaks that raise concern for H-modes at modest values of normalized beta. These arise from the plasma response at the tearing resonant surface, which is expected to depend on plasma rotation and underlying tearing stability. In experiments on DIII-D and NSTX, response and sensitivity to 3D field is found to be influenced by proximity to natural tearing stability beta limits, in addition to effects previously observed at high normalized beta or low density described by a kink-like plasma response. An increasing response develops as normalized beta rises, even from very modest values $\sim 1-2$, with required fields to induce modes falling to zero as tearing limits are approached. This response is further enhanced at low rotation and torque, with field thresholds to induce modes in torque free H-modes at normalized beta of 1.5 being well below those in Ohmic plasmas, or even plasmas above the no-wall ideal kink beta limit, where an enhanced response is expected. Both resonant ($n = 1$) and non-resonant ($n = 3$) types of field are observed to lead to modes with similar levels of applied field. An interaction with neoclassical tearing mode physics is identified, with the neoclassical modes frequently being triggered during the braking phase prior to locked mode onset. This provides the basis for a unifying criterion of locked and rotating mode onset thresholds based on braking and torque balance considerations. Thus an approach similar to Ohmic field threshold scalings can be used, and the first scalings with main parameters in torque free H-modes have been measured on this basis, identifying a more favorable density scaling but worse toroidal field scaling, and much lower thresholds in general compared to the previous Ohmic scalings on which ITER is based. Modeling of these recent results is underway with the MARS and NIMROD codes to test this understanding and confirm that the scale of effects matches current theoretical models. The results highlight fascinating new mechanisms and questions in the underlying physics, suggesting a re-evaluation of the performance and operating techniques for next step plasma regimes should be considered.

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EXS/P5-04

Macroscopic Stability of High Beta MAST Plasmas

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The high-beta capability of the spherical tokamak geometry, coupled with a suite of world-leading diagnostics on MAST, has facilitated significant improvements in the understanding of performance-limiting core instabilities in high performance plasmas. For instance, the newly installed Motional Stark Effect diagnostic, with radial resolution < 25 mm, has enabled detailed study of saturated long-lived modes (LLMs) in advanced tokamak scenarios. These modes significantly degrade confinement, damp the core rotation and enhance fast ion losses, so detailed understanding is required for their amelioration. The LLM is diagnosed as an ideal internal mode growing unstable as q_{min} approaches one and the role of rotation, fast ions and ion diamagnetic effects in determining the marginal mode stability are also discussed. Similarly, the upgraded Thomson Scattering system, with radial resolution < 10 mm and the possibility of temporal resolution of 1microsecond, has allowed detailed analysis of the density and temperature profiles in and around a Neo-classical Tearing Mode (NTM), permitting tests of models for the critical NTM island width. High resolution Charge Exchange Recombination Spectroscopy provided detailed measurement of rotation braking induced by both applied magnetic fields and by magnetohydrodynamic (MHD) instabilities, allowing tests of Neoclassical Toroidal Viscosity theory predictions. Finally, MAST is also equipped with internal and external coils that allow non-axisymmetric fields to be applied for active MHD spectroscopy of instabilities near the no-wall beta limit. The enhanced understanding of the physical mechanisms driving deleterious core MHD activity given by these leading-edge capabilities has provided guidance to optimise operating scenarios for improved plasma performance.

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EXS/P5-05

Observation of a Resistive Wall and Ferritic Wall Modes in a Line-tied, Screw Pinch Experiment

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We report an overview of theoretical and experimental results from research on the Rotating Wall Experiment, a linear screw pinch experiment at the University of Wisconsin. The overall goal of the work is to test the hypothesis that moving metal walls can stabilize the resistive wall mode (RWM) in the simplified (relative to a torus) geometry of a cylinder where spinning, conducting walls can be used instead of flowing liquid metals. The resistive wall mode has been clearly identified in the experiment; thin, thick, and ferritic shells have been used and an external kink mode has been observed whose growth rate scales inversely with the shell time. A Mumetal shell (in addition to the thick copper shell) was shown to be destabilizing – the plasma is stable without the Mumetal and destabilized with it. Finally, clear evidence for magnetic reconnection is observed in two ways, (1) in the merging of seven individual current carrying columns into a single current channel, and (2) in discrete current relaxation events which periodically flatten the current profile.

EXS/P5-06

Determination of Plasma Stability using Resonant Field Amplification in JET

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Resonant Field Amplification (RFA) has been systematically measured on JET, in two domains favourable for ITER steady-state operations: broad q-profiles with $q_{min} \sim 1$ and $q_{min} \sim 2$, at low and high β_N. MARS-F code modelling reproduces RFA data at low and high beta and suggests a new method of how the RFA data should be used to determine the no-wall limit experimentally, using logarithmic derivative of the RFA amplitude vs β_N. The modelling confirms importance of details of the current density profile on the

no-wall beta limit, rather than integral parameters (e.g. internal inductance). Simulations show that changes in the current density profile can affect the RWM beta limit if they result in destabilisation of other (i.e. current-driven peeling) modes. The nonlinear interplay between the RWM, applied magnetic fields and other plasma parameters, such as plasma rotation, has been investigated, as well as kinetic effects. A new model retaining information about the plasma response in comparison with the kinetic damping model is presented to describe the resonant field amplification in the presence of a stable RWM. Although there is no strong evidence of a beta-limit connected with the RWM on JET even at $\beta_N \sim 4$ and performance was limited by internal $n = 1$ modes, the observed (using RFA data) decrease in the no-wall limit with the increase in q_{min} is in agreement with the same dependence of the experimentally achieved highest beta values. At low beta, both $n = 1$ and $n = 2$ RFA has been observed preceding the first ELM and during ELM-free periods, and prior to appearance of low-n modes. Non-monotonic behaviour of RFA prior to the first ELM is predicted by MARS-F and observed experimentally. Stabilisation (reduction in the RFA level) due to the outer mode during ELM-free period prior to the first ELM has been found. The importance of 3D effects on stability has been considered, in particular on ballooning and global pressure-driven kink modes. 3D description of the JET wall and inclusion of 3D effects in modelling are also addressed, underlying mechanism for the observed low RFA thresholds at low betas.

Work supported by UK EPSRC and EURATOM and carried out under EFDA.

EXS/P5-07

Error Field Correction in Unstable Resistive Wall Mode Regime

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The simultaneous use of feedback control for error field correction (EFC) and stabilization of an unstable resistive wall mode (RWM) has been demonstrated in DIII-D. While the conventional EFC method addresses error fields in a pre-programmed manner, it is challenged when an unstable RWM becomes dominant, because a weakly stable or feedback-stabilized RWM becomes extremely sensitive to any small, uncorrected resonant error field. Since the DIII-D tokamak is uniquely equipped with the internal coils for fast time response and the external feedback coils for slower time response, independent magnetic feedback control in low and high frequency ranges allows the specific roles of EFC and direct feedback (DF) in active RWM control in *stable*, *marginal* and *unstable* RWM regimes. For an *unstable* RWM at the edge safety factor $q_{95} \sim 3$, the simultaneous operation of DF with the internal coils and dynamic (feedback-controlled) EFC with the

external coils enabled us not only to stabilize the unstable RWM but also to determine the necessary EFC in the presence of a feedback-stabilized RWM. However, the gain dependency of the feedback-stabilized RWM contrasts those of stable and marginal RWMs. Recent experiments and modeling show that *stable* RWMs (at $q_{95} \sim 5$ or 6) do not require high gains, while a *marginal* RWM (at $q_{95} \sim 4$) is insensitive to the feedback gains. In contrast, according to a cylindrical model developed by Okabayashi-Pomphrey-Hatcher, the EFC in the unstable RWM regime is predicted to require high gain to approach the desired correction current. This is also consistent with an on-going theoretical RWM feedback modeling using the MARS code. It has been shown that the DF bandwidth for RWM feedback control should be greater than the natural mode growth rate, consistent with theory. The established methodology to determine the optimized EFC waveform with the simultaneous use of feedback control of EFC and DF is applicable for various operational scenarios with pressure beyond the no-wall ideal stability limit. In particular, it would be highly valuable when the onset of unstable MHD is sensitive to the quality of EFC.

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EXS/P5-08

Equilibrium Reconstruction of KSTAR Plasmas with Large Uncertainty on Magnetics

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For magnetically confined plasmas such as in tokamaks and stellarators, understanding its magnetics is a basic and essential element for better physical understanding of experiments. However in KSTAR, there are two large uncertainties on magnetic measurements that make it difficult to understand and analyze the magnetics such as plasma equilibrium reconstructions. One obvious uncertainty is a ferromagnetic material (called as Incoloy-908) effect from PF and TF coils, which has non-linear effects on magnetic field generation from those coil currents. The other uncertainty is an unidentified up-down asymmetric field source. Due to this unidentified source, all KSTAR plasmas since the first one in 2008 were produced around $Z \sim -10$ cm below the mid-plane, even though applied PF coil currents were up-down symmetric. As a result, it turned out that an appropriate external field source modeling is essential for robust and reliable equilibrium reconstruction of KSTAR plasmas. Therefore a new magnetic analysis code, named as IDK, for robust equilibrium reconstructions under those large uncertainties was developed with an external field source modeling. For its validations, it was applied to equilibrium reconstruction of KSTAR plasmas and confirmed quantitatively by comparisons with TV (CCD) image analysis. Also it was applied to a basic stability analysis such as vertical and radial plasma instabilities to understand the characteristics of discharge terminations for 2009 campaign, so that it turned out that most of disruptive discharge terminations were due to the vertical displacement event (VDE). With emphasis on ferromagnetic materia's effect, it was applied to a vacuum field analysis to understand the effect on magnetic

configurations for plasma startup. The comparison with predictions by a 2D FEM Incoloy model showed good agreements and important implications for future experiments of KSTAR.

EXS/P5-09

Non linear MHD Modelling of NTMs in JET Advanced Scenarios

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In Advanced Tokamak Scenarios, with $q_{min} > 1$, the $(2, 1)$ NTM is one major MHD limit. We have investigated its non linear threshold on a JET case using the non linear MHD code XTOR. This study addresses several points, related to theory (comparison with Rutherford) and comparison to experiment (threshold dynamics, confinement degradation, mode structure). We find that the Extended Rutherford equation does not match very well non linear simulations except when the curvature term becomes dominant, due to the limitations of Delta' models. It appears also that the role of magnetic shear on the threshold explains its decay during current diffusion, due to the weakening of the curvature stabilization. Thus for a given seed amplitude, NTM triggering becomes easier as the current diffuses. The dynamics of the confinement degradation associated with the slow growth of the island is comparable to experimental observations, and mode structure, reconstructed from a synthetic ECE diagnostic, is similar in shape, although differences in linear and non linear coupling to adjacent modes exist. We also address the role of toroidal rotation on the threshold, which has been evidenced experimentally on several devices, showing an increase of the β_N threshold with rotation. Surprisingly, we find that the effect of rotation is slightly destabilizing for $P_{rm} = 10$, and insignificant for $P_{rm} = 100$, where P_{rm} is the Prandtl number. These values correspond approximately to the theoretical collisional perpendicular viscosity and to the toroidal viscosity from momentum balance, respectively. In this regime, the intrinsic NTM stability is therefore not significantly affected by rotation, in contrast with previous experimental indications. The rotation scaling could be more sensitive to plasma equilibrium (the database is established from $q < 1$ high performance discharges), or the role of the primary mode (and its coupling) could be more important, than expected.

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EXS/P5-10

Three-Dimensional Physics Studies in RFX-Mod

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The discovery of Single Helical Axis States (SHAx) in the reversed field pinch (RFP) gives a unique opportunity to investigate the physics of three-dimensional fields in magnetized fusion plasmas. In RFX-mod high current experiments the plasma frequently reaches the Single Helical Axis (SHAx) state, in which the core electron temperature assumes a helical shape and significant gradients appear. The plasma magnetic topology is strongly helical in the core while it is almost axisymmetric at the edge, thanks to the active control of the edge radial field. The weak helical field produced by the plasma profoundly affects the shape of the equilibrium going from 2D to 3D, therefore three-dimensional tools are being developed for the description of these states. On one hand the SHEq code, based on a perturbative approach, is used to determine the magnetic field topology by computing the helical flux: the iota profile of the helical flux surfaces significantly differs from the axisymmetric one especially in the core. On the other hand, the VMEC code has been modified for the helical RFP, in order to describe the SHAx states by using tools developed by the Stellarator community. The VMEC code allows deepening the investigations on the use of helical fields to allow long pulse operations. Moreover, an investigation on the stability of the helical states is being performed by means of the TERPSICHORE code: preliminary runs suggest that the core magnetic core shear is an important parameter for ideal MHD stability. The effect of three-dimensional fields on transport is also being actively investigated. The magnetic shear is found to be correlated with regions with reduced transport: an investigation on the transport mechanisms underlying such barriers is being performed by adopting several approaches. Particle transport coefficients across helical surfaces have been estimated numerically by a mono-energetic test particle approach: the main contribution to transport is found to be due to neoclassical effects, even though a residual level of magnetic chaos may play a role on passing particles. Further investigations at low collisionality show that the estimated particle transport coefficients do decrease in these regimes. A different approach for transport characterization based on Stellarator tools, using VMEC or SHEq equilibria, is planned.

EXS/P5-11

Mode Structure of Global MHD Instabilities and its Effect on Plasma Confinement in LHD

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We have performed quantitative evaluation of the effect of the peripheral $m = 1/n = 1$ mode on the plasma confinement in LHD. Accumulation of experimental data on the relation between the mode structure and edge magnetic fluctuation is also in progress. The LHD magnetic field configuration is characterized by high magnetic shear with low magnetic well or magnetic hill configuration particularly at the plasma periphery. This characteristic leads to persistent pressure-driven edge resonant MHD instabilities. The mode structure was estimated from $m = 1/n = 1$ soft-X ray (SXR) fluctuation. The line-integrated SXR fluctuation profile was converted to the radial profile using the Gaussian profile model peaked at the resonant surface, and the experimental mode width Δ_{sx} was defined from this profile. It is confirmed that the peripheral $m = 1/n = 1$ mode with normalized width of 5% has caused degradation of energy confinement by 10%. The $m = 1/n = 1$ edge magnetic fluctuation level is 0.01% when the mode with 5% width exists. In the range of magnetic fluctuation level lower than 0.01%, we have compared Δ_{sx} with magnetic island width Δ_{cs} due to virtual tearing-type current driven mode which would produce the experimental magnetic fluctuation at the edge. The results show that Δ_{sx}'s are about 1/3 of the Δ_{cs}'s. The discrepancy characterizes the pressure driven peripheral MHD instability. Accumulation of the data would make it possible to compare them with theoretical predictions of mode width of the pressure driven peripheral MHD instability.

EXS/P5-12

Robust Correction of 3D Error Fields in Tokamaks including ITER

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Important progress has been made in the correction of 3D fields, based on the improved understanding of plasma response using the Ideal Perturbed Equilibrium Code (IPEC) [1-3]. The key to error field correction is to reduce the dominant distribution of 3D fields that is stronger often by an order of magnitude than any other distribution in breaking magnetic surfaces. The important validation is achieved in presently the most extreme case, the DIII-D mock-up experiments for the ITER Test Blanket Modules (TBMs). Although the TBM 3D fields are highly localized and can not be controlled by typical error field correction coils, the optimal operation could be restored using I-coils by minimizing the dominant part in the TBM 3D fields as IPEC prediction. Including TBM experiments, various experimental results in error field correction and locking in tokamaks have been successfully understood and quantified based on the dominant 3D fields, as can be summarized in the robust parametric scaling of the locking threshold across many different plasma configurations. The implications are favorable for ITER, since the highly reliable 3D field compensation can be provided for a wide range of different plasmas if the correction coil is designed based on the robust patterns of the dominant 3D fields. Present Error Field Correction Coil (EFCC) in ITER is under active investigations using IPEC to revise their capabilities [4].

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EXS/P5-13

Exploration of Optimal High-beta Operation Regime by Magnetic Axis Swing in the Large Helical Device

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In Large Helical Device (LHD), the volome averaged beta value as high as 5.1% was achieved in FY2007-2008 experiments. High beta operation regime was explorated by the programmed control of magnetic axis position, which characterizes MHD equilibrium, stability and transport. This control became enable by increasing capability of poloidal coil power supply. The experiments made clear the effect of magnetic hill on MHD activities in high-beta plasmas with more than 4%. Also it enabled to access the ideal stability boundary with keeping high-beta state. The strong $m/n = 2/1$ mode leading minor collapse in core plasma appeared with the inward shift of the magnetic axis.

EXS/P5-14

Characteristics of Extremely Deep Reversal and Quasi-Single-Helicity (QSH) States in a Low-aspect-ratio RFP

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In a low-aspect-ratio RFP machine RELAX ($A = R/a = 0.5\,\mathrm{m}/0.25\,\mathrm{m}$), two important discharge regions in (Theta, F) space have been realized. The discharge and plasma parameters in RELAX are as follows: plasma current of up to 100 kA, discharge duration of up to 2.5 ms, electron density around 10^{19} m^{-3} using interferometry, and electron temperature $<$ 100 eV from soft-X ray (SXR) measurement. In RELAX discharges, extremely high-Theta (> 3), deep-reversal ($F < -1$) region can be attained without sawtooth crash or discrete dynamo event unlike the conventional RFPs. The q profile from equilibrium reconstruction shows strong magnetic shear in the outer region. The shear may result in lower mode amplitudes with broad toroidal mode spectrum of the magnetic fluctuation. The toroidal mode spectrum is narrowed by reducing the field reversal, to that of the quasi-single helicity (QSH) state in very shallow reversal discharges. In shallow reversal region, SXR pin-hole images and multi-chord SXR measurements of such QSH state have indicated helical hot core of either a large magnetic island or helically deformed core. We will discuss these characteristics of low-A RFP from the viewpoint of core plasma for a compact fusion reactor.

EXS/P5-15

Control of MHD Instabilities in the STOR-M Tokamak using Resonant Helical Coils

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Previous studies on the MHD properties in the STOR-M tokamak ($R/a = 0.46/0.12$ cm, $B_t = 1$ T, $I_p < 50$ kA) have shown that the dominant $m = 2$ instabilities and their interplay with other modes played a vital role in the H-mode like discharges triggered by a short current pulse, electrode biasing and compact torus injection, both for L-H transition and H-L back transition. Present studies concentrate on active control of the $m = 2$ instabilities using a set of resonant helical coils (RHCs) in the $m/n = 2/1$ configuration. Numerical simulation based on the STOR-M configuration indicates that a magnetic island's width decreases with increasing RHC current until it completely disappears and the island reappears and grows with further increases in the RHC current, consistent with the previous experimental observations which indicated MHD stabilization for moderate RHC current magnitude and destabilization for further increased RHC current. When an 800 A RHC current is applied during a 25 kA STOR-M discharge, significant reduction in magnetic fluctuation amplitude, particularly for the $m = 2$ mode, and change in frequency characteristics have been observed. Significant reduction ($\sim 40\%$) in hydrogen line emission level and increase in global energy confinement time have also been observed.

EXS/P7 — Poster Session

EXS/P7-01

Studies of MHD Effects on Fast Ions: towards Burning Plasma with ITER-like Wall on JET

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Fast ion studies on JET aiming at assessing the peak values of fast-ion losses caused by plasma disruptions, TAE, and NTM are performed. This work is carried out as continuation of fast ion research on JET due to its significance for the future JET operation with ITER-like Be wall. Study of the MHD induced peak losses of fast ions play a crucial role for the PFC selection for DEMO. Gamma-ray diagnostics, neutron spectrometry, NPA, Faraday Cups and Scintillator Probe (SP) were used for simultaneous measurements of various species of confined and lost fast ions in the MeV energy range. The high time resolution of diagnostics allowed studying both resonant and non-resonant MHD effects

on redistribution and losses of energetic ions. The fast ion populations were generated in fusion reactions and were also produced by NBI and by accelerating with ICRH. Significant fast ion losses preceding disruptions were often detected by SP in discharges with high β_N. It was found that the losses are caused by the kink mode and these losses typically occur at the same time as the thermal quench before the current quench that follows. Bursts of metal impurities influx following the losses were observed. A set of experiments were carried out where interactions of core-localised TAE modes with fast ions in the MeV energy range were studied in plasmas with monster sawteeth. Simultaneous measurements of confined and lost ions allowed distinguishing the energy ranges of fast ions and their spatial redistribution during the TAE activity preceding monster sawtooth crashes. Energy and pitch angle resolved SP measurements of MeV-ions ejected from the plasma due to the non-resonant fishbone oscillations driven by NBI-ions are also studied. The lost ions are identified as fast protons accelerated by ICRH. Numerical simulations were performed with HAGIS, MISHKA and SELFO codes. Experiments with EFCC stabilising ELMs have exhibited a strong effect of EFCC on NTM and on NTM-induced losses of NBI ions. It was found that amplitude of NTMs decreases significantly during EFCC while the frequency of NTM sweeps down. The decrease in the NTM amplitude has a profound effect on losses.

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EXS/P7-02

Theoretical and Experimental Analysis of the Destabilization of Modes by Fast Particles in Tore-Supra

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Suprathermal particles are well known to excite collective modes, which can have deleterious effects on their confinement. It is important to determine the mechanisms and conditions of this excitation to minimize losses of suprathermal particles. We report on such stability studies, based on comparisons between theory and experiments carried out on Tore-Supra for two low frequency modes: fishbone modes driven by Lower Hybrid induced fast electrons [1] and Beta Alfvén Eigenmodes (BAEs) driven by Ion Cyclotron Heating induced fast ions [2].

Signatures of electron fishbone modes have been identified on both temperature fluctuation and hard-X rays diagnostics, with an unexpected jumping behavior of the modes frequency [1]. We conducted the linear stability analysis of the modes using the linear solver MIKE coupled with a reconstruction of the suprathermal electron population with the Fokker-Planck C3PO/LUKE code. The analysis displays an important role of the mode MHD structure and the necessity of a q-profile inversion for the mode drive.

BAEs are acoustic modes whose dispersion properties and structure are known to depend strongly on kinetic effects [2, 3]. An approximate linear calculation of the impact of global macroscopic parameters in the BAE stability, making use of a perturbative treatment of Landau damping and fast particle drive, confirmed the possibility of the mode destabilization by resonant interaction with suprathermal ions in Tore-Supra [2]. Here, two directions are investigated to improve the stability analysis. First, we show the importance of inertial kinetic effects in the mode frequency and stability, supported by the experimental BAE dynamics observed during a sawtooth period. Secondly, we study the possibility of a nonlinear modification of the BAE stability, due to the nonlinear trapping of resonant particles inside the BAE structure [4]. We show that nonlinear trapping can change the magnitude of the driving and damping mechanisms at stake and lead to the existence of metastable modes.

[1] A. Macor et al., PRL 102, 155005 (2009) / J. Decker et al., EFTC Riga (2009).

[2] C. Nguyen et al., PoP 15, 112502 (2008) / R. Sabot et al., NF 49, 085033 (2009) / C. Nguyen et al., PPCF 51, 095002 (2009).

[3] F. Zonca et al., PoP 6, 1917 (1999) / Ph. Lauber et al., IAEA-TCM-EP Kiev (2009).

[4] H.-L. Berk et al., PFB 2, 9 (1990).

EXS/P8 — Poster Session

EXS/P8-01

Long-range Correlations and Impurity and MHD Effects on Saw-tooth Oscillation in TCABR Tokamak Biasing and Alfvén Heating Experiments

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Results of an experimental campaign in TCABR, organized within the framework of the IAEA Coordinated Research Project on "Joint Experiments Using Small Tokamaks" (May 2009), are reported. The experiments were carried out with the objective of investigating fluctuation and transport phenomena in improved confinement discharges triggered by external electrostatic biasing together with low-power Alfvén wave heating. It is shown that long distance correlations are observed in the fluctuating electrostatic potential at the plasma edge and that central impurity accumulation and reduction of the amplitude of saw-tooth oscillations are observed during Alfvén heating.

EXS/P8-02

Experimental Study of Poloidal Flow Effect on Magnetic Island Dynamics in LHD and TJ-II

Narushima, Y.[1], Castejón, F.[2], Sakakibara, S.[1], Watanabe, K.Y.[1], Ohdachi, S.[1], Suzuki, Y.[1], Estrada, T.[2], Medina, F.[2], Lopez-Bruna, D.[2], Yokoyama, M.[1], Yoshinuma, M.[1], Ida, K.[1], LHD Experiment Group[1], and TJ-II Experiment Group[2]

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The dynamics of a magnetic island are studied by focusing on the poloidal flows in the helical devices LHD and TJ-II. The intended magnetic islands are non-rotating in both devices. The temporal increment of the $\mathbf{E} \times \mathbf{B}$ poloidal flow prior to the magnetic island transition from growth to healing and the modification of the current sheet in the plasma during the transition are observed. These experimental observations mean that the poloidal flow affects the magnetic island dynamics. In LHD experiments, the electron-diamagnetic directed poloidal flow outside the rational surface, $\iota = 1$, increases prior to the magnetic island (poloidal/toroidal Fourier mode number of $m/n = 1/1$) being healed. The island width decreases after the poloidal flow increases. Therefore, there exists a threshold of poloidal flow for healing. From the magnetic diagnostics, it is observed that the structure of a current sheet flowing in the plasma moves about 180-degree poloidally in the electron-diamagnetic direction during the transition. A similar behavior of magnetic island dynamics is also observed in TJ-II plasmas despite the different parameter range (mode number of island $m/n = 2/4$, lower magnetic shear, lower-beta, lower-collisionality, and ion-diamagnetic direction of the poloidal flow etc.) from LHD. Island healing follows the increase of the ion-diamagnetic directed poloidal flow at the rational surface $\iota = 2$. Before the increase of the poloidal flow, the magnetic island with $m/n = 2/4$ exists and lasts until just after the increase of the poloidal flow. Afterward, the magnetic island disappears. The experimental observation in TJ-II shows that the poloidal flow changes prior to the island dynamics (transition) as in LHD. These experimental observations from LHD and TJ-II show that the temporal increment of the poloidal flow is followed by the transition (growth to healing) of the magnetic island regardless of the flow direction and clarify the fact that significant poloidal flow affects the magnetic island dynamics. It is thought that due to the increase of the poloidal flow, the viscous drag force overcomes the magnetic torque between the externally imposed field and the current sheet. As a result, the current sheet is shifted (rotated) and heals the magnetic island.

EXW — Oral

(*Magnetic Confinement Experiments: Wave-plasma interactions - current drive, heating, energetic particles*)

EXW/4-1

ICRF Mode Conversion Flow Drive on Alcator C-Mod

Lin, Y.[1], Rice, J.E.[1], Wukitch, S.J.[1], Reinke, M.L.[1], Greenwald, M.[1], Hubbard, A.E.[1], Marmar, E.S.[1], Podpaly, Y.[1], Porkolab, M.[1], Tsujii, N.[1], and Alcator C-Mod Team[1]

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ICRF mode conversion flow drive (MCFD) may be a candidate for the external control of plasma rotation in large tokamaks like ITER [1]. Recently, we have carried out a detailed study on MCFD, including its dependence on plasma and RF parameters. These results shed some light on the underlying physics and may help to extrapolate the method to other fusion devices. The observed change in the toroidal rotation (dV) is always in the co-I_p direction. The flow drive efficiency is found to depend strongly on the ^{3}He concentration in D(^{3}He) plasmas, a key parameter separating the ICRF minority heating and mode conversion regimes. This result further supports the key role of mode conversion. The efficiency is also strongly affected by plasma density ($\sim 1/n_e$), i.e., a power and/or momentum per particle dependence. The flow drive efficiency at $f = 78$ MHz and $B_t(0) = 8$ T is lower than that at 50 MHz/5.1 T, possibly due to the $1/f$ dependence of wave momentum at the same RF power. The change of rotation also increases with I_p, indicating momentum confinement dependence. At +90 degree antenna phase (waves in co-I_p direction) and dipole phase (waves symmetrical in both directions), we find that dV is proportional to the RF power up to the maximum available RF power. A central $dV \sim 110$ km/s in low density L-mode has been achieved. The rotation at -90 degree antenna phase is usually smaller, and the flow drive effect appears to be saturated (or decreased) at high RF power. The difference vs. antenna phases indicates that possibly two mechanisms are involved in determining the total flow drive torque: one is RF power dependent and generates a torque in the co-I_p direction, and the other is wave momentum dependent, i.e., the torque changes direction vs. antenna phase. Results in H-mode follow the same parametric scaling. However, the observed change of rotation in H-mode has been small because of the much higher density and the unfavorable $1/n_e$ scaling.

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EXW/4-2

Transport of Energetic Ions Due to Microturbulence, Sawteeth, and Alfven Eigenmodes

Pace, D.C.[1], Fisher, R.K.[2], García-Muñoz, M.[3], Heidbrink, W.W.[1], Lin, Z.[1], McKee, G.R.[4], Murakami, M.[5], Muscatello, C.[1], Nazikian, R.[6], Park, J.M.[5], Petty, C.C.[2], Rhodes, T.L.[7], Van Zeeland, M.A.[2], Waltz, R.E.[2], White, R.B.[6], Yu, J.H.[8], Zhang, W.[1], and Zhu, Y.B.[1]

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Experimental results from the DIII-D tokamak, coupled with advanced theoretical treatments, elucidate a variety of energetic ion transport behaviors. A crucial issue for fusion reactors is the confinement of 3.5 MeV fusion born alpha particles that must deposit their energy in the background plasma in order to maintain a burning state. Once these particles have deposited their energy, the challenge is to remove this alpha ash before it begins to absorb energy intended for the deuterium/tritium fuel. Instabilities ranging from large-scale sawteeth to fine scale microturbulence significantly affect the confinement of alphas and the transport of helium ash. Consequently, validating predictive models for this non-linear interaction is vital for extrapolation to ITER and beyond. This non-classical fast ion transport is investigated in the DIII-D tokamak using an array of new diagnostics and simulation/modeling techniques. For transport by microturbulence, experiments compare observed energetic ion behavior to expectations based on classical (i.e., collisional) theory. It is found that the difference between observation and theory increases as the ratio T/E increases (T is the plasma temperature and E is the energy of the energetic ions) [1,2]. This behavior is predicted by theoretical treatments [3-5]. These findings indicate that microturbulence will likely increase the transport of alpha ash ($T/E \sim 1$) while having little to no effect on the confinement of fusion alphas ($T/E \ll 1$). In plasmas featuring sawtooth events similar to those of the ITER baseline scenario, major redistribution of the energetic ion population is observed by fast-ion D_α and loss detector diagnostics. Finally, high transport levels in plasmas featuring Alfvén eigenmodes [6] are accurately described by theory that considers the integrated effect of a large number of simultaneous modes [7].

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[2] W.W. Heidbrink et al., Plasma Phys. Control. Fusion **51**, 125001 (2009)

[3] C. Estrada-Mila et al., Phys. Plasmas 13, 112303 (2006)

[4] W. Zhang et al., Phys. Rev. Lett. 101, 095001 (2008)

[5] T. Hauff et al., Phys. Rev. Lett. 102, 075004 (2009)

[6] W.W. Heidbrink et al., Nucl. Fusion 48, 084001 (2008)

[7] R. White et al., Phys. Plasmas 17 (2010) in press.

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EXW/4-3Rb

Potential Fluctuation Associated with Energetic-Particle Induced Geodesic Acoustic Mode in Reversed Magnetic Shear Plasmas on LHD

Ido, T.[1], Shimizu, A.[1], Nishiura, M.[1], Nakamura, S.[1], Kato, S.[1], Nakano, H.[1], Yoshimura, Y.[1], Toi, K.[1], Ida, K.[1], Yoshinuma, M.[1], Satake, S.[1], Watanabe, F.[2], Kubo, S.[1], Shimozuma, T.[1], Igami, H.[1], Takahashi, H.[1], Yamada, I.[1], Narihara, K.[1], and LHD Experiment Group[1]

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Characteristics of energetic-particle induced geodesic-acoustic-mode(GAM) were investigated through direct and local measurement of the electrostatic potential fluctuation, which is an essential physical quantity in the GAM, the oscillatory zonal flow, by using a heavy ion beam probe in the LHD plasmas. The GAM localizes near the central region of the plasma. It is also confirmed that the mode possess the oscillatory radial electric field which induces the oscillatory $\mathbf{E} \times \mathbf{B}$ flow.

EXW/4-4Rb

Destabilization of Beta-induced Alfvén Eigenmodes in the HL-2A Tokamak

Chen, W.[1], Liu, Yi[1], Yang, Q.W.[1], Ding, X.T.[1], Ji, X.Q.[1], Zhang, Y.P.[1], Dong, Y.B.[1], Huang, Y.[1], Zhou, J.[1], Zhou, Y.[1], Li, W.[1], Song, X.M.[1], Dong, J.Q.[1], Cao, J.Y.[1], Shi, Z.B.[1], Duan, X.R.[1], and HL-2A Team[1]

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The beta-induced Alfvén eigenmodes (BAEs) were first observed in DIII-D and TFTR plasmas with fast ions. Subsequently, the BAEs have also been observed during a stronger tearing mode (TM) activity in FTU and TEXTOR Ohmic plasmas without fast ions. Recently, the BAEs have also been reported during a sawtooth cycle in ASDEX-U and TORE-SUPRA plasmas with fast ions To date, the most probable theoretic identifications of the BAEs excited by energetic particles and magnetic island have been proposed: a discrete shear Alfvén eigenmode (AE), a kinetic ballooning mode (KBM) and a hybrid mode between Alfvénic and KBM branches. The excitation mechanism of the BAEs is mysterious due to many kinetic effects, such as diamagnetic drift, thermal ion compression, finite Larmor radius and temperature/density gradient.

In this paper, it is reported for the first time that the experimental results are associated with the BAE during ECRH, and it is also presented that ones are correspond to the BAE during a strong tearing mode activity and that in Ohmic plasma without large magnetic island on HL-2A. After comparing with these three cases, it has been found that the mode numbers of the BAEs excited by magnetic island are $m/n = 2/1$ and $-2/-1$, and the BAEs propagate poloidally and toroidally in opposite directions, and form standing-wave structures in the island rest frame, and there is an island width threshold of the BAEs excitation. However, the mode numbers of the BAEs in ECRH plasma and Ohmic plasma without large magnetic island are $m/n = 3/1$, and the BAEs all propagate poloidally in electron diamagnetic direction. The frequencies ($f = 10 - 30$ kHz) of the BAEs are all proportional to Alfvén velocity, and the frequencies of the BAEs excited by energetic electrons are proportional to the average energetic electron beta, and the frequencies of the BAEs driven by magnetic islands are proportional to the island width. The island width threshold of BAE excitation has been found using a statistical analysis method. The threshold is about 3.4 cm on HL-2A. In addition, an interesting result about the BAEs modulated by supersonic molecular beam (SMB) and gas puffing has been present in the paper.

EXW/10-2Ra

Avoidance of Disruptions at High β_N in ASDEX Upgrade with off-Axis ECRH

Esposito, B.[1], Granucci, G.[2], Maraschek, M.[3], Gude, A.[3], Igochine, V.[3], Nowak, S.[2], Stober, J.[3], Treutterer, W.[3], Zohm, H.[3], and ASDEX Upgrade Team

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The control of disruptions in tokamak plasmas by means of electron cyclotron resonance heating (ECRH) has been addressed in several machines. The technique is based on the stabilization of magnetohydrodynamic (MHD) modes occurring at a disruption through the localized injection of ECRH on a resonant surface. A delay in the occurrence of disruption has been achieved in some cases and complete avoidance in other cases. So far, these experiments have dealt with L-mode plasmas. The first experiments of this type carried out in H-mode plasmas are described in this paper. Delay and/or complete avoidance of disruptions has been achieved in ASDEX Upgrade using localized injection of ECRH (1.5 MW) in a high β_N scenario (1 MA, 2.2 T, with NBI (7.5 MW)). In these discharges (at relatively low q_{95} and low density) neoclassical tearing modes (NTMs) are excited and when they lock a disruption occurs. The same scheme used in previous disruption avoidance experiments in FTU and AUG has been applied: as soon as the disruption precursor signal (loop voltage and/or locked mode detector) reaches the preset threshold, the ECRH power is triggered by real time control. A poloidal scan in deposition location has been performed by varying the poloidal angles of the launching mirrors in different discharges. Complete avoidance is achieved when the power is injected close to the $q = 3/2$ surface: in this case multiple unlocking of MHD modes occurs after ECRH application. When ECRH is injected at more external locations, the discharges, although not disrupting immediately as in the reference case, show no mode unlocking and eventually disrupt. For injection at inner locations the discharge disrupts as in the reference case. The absorption of ECRH is therefore found to modify the sequence of events occurring at a disruption by acting locally (through a change in resistivity) on the gradient of plasma current which is supposed to be the main drive of the island amplitude growth rate.

EXW/P2 — Poster Session

EXW/P2-01

Control of MHD Modes with a Line-of-Sight ECE Diagnostic

de Baar, M.R.[1], Bongers, W.A.[1], Bürger, A.[3], Donné, A.J.H.[1,2], Hennen, B.A.[1,2], Nuij, P.W.J.M.[2], Oosterbeek, J.W.[2,3], Westerhof, E.[1], Witvoet, G.[2], Steinbuch, M.[2], Thoen, D.J.[1], and TEXTOR Team

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The experimental proof-of-principle of a feedback control approach for realtime, autonomous control of tearing modes and the sawtooth period in tokamaks is reported. The control systems are designed using a control systems engineering approach and combines an Electron Cyclotron Emission (ECE) diagnostic for sensing of the tearing modes in the same sight-line with a steerable Electron Cyclotron Resonance Heating and Current Drive (ECRH/ECCD) antenna. Methods for fast detection of $q = m/n = 2/1$ tearing modes and accurate retrieval of their location, rotation frequency and phase will be presented. Subsequently, the set-points to establish alignment of the ECRH/ECCD deposition location with the centre of the tearing modes are derived in real-time and forwarded to the real-time control system to actuate the ECRH/ECCD launcher and gyrotron. The design of the control system and the controllers used therein are explained.

EXW/P2-02

Non-inductive Plasma Current Start-up Experiments in the TST-2 Spherical Tokamak

Ejiri, A.[1], Kurashina, H.[1], Takase, Y.[1], Hanashima, K.[1], Sakamoto, T.[1], Watanabe, O.[1], Nagashima, Y.[1], Yamaguchi, T.[1], An, B.I.[1], Kobayashi, H.[1], Hayashi, H.[1], Yamada, K.[1], Kakuda, H.[1], Hiratsuka, J.[1], Wakatsuki, T.[1], and Goto, M.[2]

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Non-inductive plasma start-up experiments have been performed on the TST-2 spherical tokamak device using ECH (2.45 GHz/ 5 kW). Core equilibrium information during the current sustained phase was extracted combining various diagnostics and an equilibrium calculation with anisotropic pressure. The analysis indicates that a banana-shaped high density and high temperature region is located in the outboard region. This result emphasizes the significance of trapped electrons.

The ECH power was modulated to see various responses. The response of the visible CCD camera images indicate that the core electron temperature is in the burn through phase of light impurity radiation, and the peripheral faint region has a higher temperature than the core. According to the Thomson scattering measurement, the central ($R = 400$ mm) temperature is about $5-10$ eV, while probe measurements show that the edge temperature is about 15 eV. In addition, the HeI line intensity ratio (706 nm/728 nm) measurement in deuterium-helium mixed plasma suggests higher electron temperature in the outer peripheral region.

Line densities along five chords are measured by a 50 GHz microwave interferometer. Using the measured line densities, and assuming a flat density profile, we reconstructed the shape of the high density region. The resultant shape looks like a banana placed along the outboard boundary, and the line averaged density is above the cutoff density for the EC wave. A new equilibrium code including anisotropic pressure was developed. We found an equilibrium, which is consistent with the external magnetic measurements, and which is characterized by an outboard banana-shaped high current density and high pressure region.

The sustainment of the banana-shaped high density, high temperature region can be interpreted by EC wave refraction and the consequent poor wave accessibility to the core region. Once the high density banana-shaped region is formed, microwave launched from the weak field side cannot reach the core region, leading to the sustainment of this structure. Finally, the electrons are marginally collisionless, and hence the equilibrium and the current drive mechanism are probably determined by trapped electrons.

EXW/P2-03

Plasma Start-up Results with EC Assisted Breakdown on FTU

Granucci, G.[1], Ramponi, G.[1], Calabrò, G.[2], Crisanti, F.[2], Ramogida, G.[2], Bin, W.[1], Botrugno, A.[2], Buratti, P.[2], D'Arcangelo, O.[1], Frigione, D.[2], Pucella, G.[2], Romano, A.[2], Tudisco, O.[2], and FTU2 and ECRH1 Team

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Several experiments aimed at optimizing plasma pre-ionization by using EC (Electron Cyclotron) waves have been carried out on many tokamak in recent years as the basis of a multi-machine comparison study made to define the best operation scenarios for ITER. The pre-formation of a low density plasma allows the start-up of plasma current with a reduced electric field as is foreseen for ITER where the plasma break down will have to be achieved with a toroidal electric field of only 0.3 V/m. The FTU (Frascati Tokamak Upgrade, $R = 0.935$ m, $a = 0.3$ m) contribution to this study is the main subject of the present work. Moreover a reduction in electric field and internal inductance (li), as can be obtained with pre-ionization by ECH, can lower the transformer flux consumption in the start-up phase leading to a longer plasma current flat top. This point is of particular attention in the conceptual design of the steady state scenario of the proposed FAST tokamak and has been addressed too. The FTU experimental set up is based on the use of two gyrotrons (140 GHz, up to 0.8 MW) switched on at the start of plasma sequence ($t = 0$ s) and injecting the power for fundamental ordinary mode polarization (O1). The magnetic field has been varied in the range $4.6 - 5.8$ T while perpendicular and oblique toroidal injection (20°) of EC wave has been compared. A scan in pre-filling pressure has evidenced the capability of EC power to increase, by a factor 4, the range of working pressure useful for plasma start-up. Varying the breakdown electric field a minimum of 0.37 V/m has been found with 0.8 MW of EC in perpendicular injection. A toroidal launching angle of 20° exhibits a reduced efficiency needing a factor 2 in power to obtain same results. The scan in resonance position has demonstrated that a reproducible control on plasma start up location can be easily obtained with EC pre-ionization. A total transformer flux saving of 22% has been found acting on plasma resistivity (by increasing electron temperature) and on plasma starting point (for an internal inductance reduction).

EXW/P2-04

Power Supply of Vertical Stability Coil in EAST

Huang, H.H.[1], Gao, G.[1], Liu, Z.Z.[1], Fu, P.[1], Wu, Y.B.[1], Liu, X.Y.[1], Fang, L.[1], Fang, H.X.[2], and Hu, X.Q.[2]

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Power supply of vertical stability coil in EAST(Experimental Advanced Superconducting Tokamak) is a large capacity single phase inverter power supply, which traces displacement signal of plasma, and excites four fast-control coils in vacuum chamber to produce magnetic field that realizes plasma stabilization in large elongate model. It consists of HV switches, AC/DC converters, 24 IGBT modules of 3-level half bridges with diodes neutral point clamping in parallel and control to meet the requirement of large current and fast response. The technique of carrier wave Phase-shift PWM is applied in IGBT modules to decrease switching loss of IGBT, raise equal switch-frequency of converter and improve performance of output wave. The validity of proposed scheme and control strategy were confirmed by simulation and experiments. It has been under operation since 2006 in the EAST.

EXW/P2-05

ECH-assisted Startup using Pre-ionization by the Second Harmonic 84 GHz and 110 GHz EC Waves in KSTAR

Joung, M.[1], Bae, Y.S.[1], Park, S.I.[2], Jeong, J.H.[2], Han, W.S.[1], Kim, J.S.[1], Do, H.J.[2], Yang, H.L.[1], Namkung, W.[2], Cho, M.H.[2], Humphreys, D.[3], Walker, M.L.[3], Leuer, J.A.[3], Hyatt, A.W.[3], Eidietis, N.W.[3], Gorelov, Y.[3], Lohr, J.[3], Doane, J.[3], Mueller, D.[4], and KSTAR Team

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The second harmonic electron cyclotron heating (ECH)-assisted startup has been established to provide the reliable plasma startup with low toroidal loop voltage in the Korean Superconducting Tokamak Advanced Research (KSTAR) which is a fully superconducting tokamak device in Korea. The experimental results during KSTAR 2008 and 2009 campaigns showed the feasibility of the second harmonic 84 and 110 GHz ECH-assisted startup with the low loop voltage ranged from 3 V (0.26 V/m) to 4 V (0.35 V/m). The application of the EC beam before the Ohmic provided a localized plasma, which is called as 'pre-ionization' as providing many hot electrons and thereby reduction of the resistive power consumption of the Ohmic flux. It also allowed the burn-through and sustained the plasma during the current ramp-up. The interesting thing is that the pre-ionization failed

with a pure toroidal field and no poloidal magnetic field null (FN) structure. Also, the small amount ($\sim$ 1 kA) of the toroidal plasma current was observed in the pre-ionization phase. It is considered that the vertical field component of the FN structure plays something role of confining the electrons during the pre-ionization phase. During the 2009 plasma campaign, the optimized condition of ECH pre-ionization was investigated with parameter scans of hydrogen and deuterium pre-fill gas pressure, resonance position, polarization, and vertical magnetic field without Ohmic discharge. In the KSTAR 2010 campaign, the interrelationship between the reliable and reproducible plasma discharge in Ohmic phase and the ECH pre-ionization is studied experimentally with the second harmonic 110 GHz EC beam at the upgraded power up to $\sim$ 400 kW. Also, the fundamental harmonic 84 GHz ECH-assisted startup is attempted and its results are compared with those of the second harmonic 110 GHz ECH-assisted startup. This paper presents the overview of the startup ECH system and experimental results of 84 GHz and 110 GHz ECH-assisted startup from the first plasma campaign in 2008 to the 2010 campaign.

EXW/P2-06

Confinement Measurements and MHD Simulations of Oscillating-Field Current Drive in a Reversed-Field Pinch

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Oscillating-field current drive (OFCD) is a method of steady-state plasma sustainment proposed for the reversed-field pinch (RFP) wherein poloidal and toroidal AC loop voltages are applied to produce a DC toroidal plasma current with magnetic relaxation. OFCD has been added to RFPs sustained by standard toroidal induction in order to increase the plasma current by up to about 10% in the MST device. Recently these experiments have been numerically modeled with nonlinear 3D resistive MHD, and the calculated dependence of the added current on the phase between the two AC voltages agrees with experiment. Also, the recently measured energy confinement in the optimum OFCD case is slightly improved compared to that in the standard RFP.

EXW/P2-07

Plasma Models for Real-Time Control of Advanced Tokamak Scenarios

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An integrated plasma profile control strategy, ARTAEMIS, is being developed for extrapolating present-day advanced tokamak (AT) scenarios to steady state operation. The approach is based on semi-empirical modeling. It was initially explored on JET, for current profile control only [1]. The present paper deals with the generalization of this strategy to simultaneous magnetic and kinetic control. The methodology is generic and can be applied to any device, with different sets of heating and current drive actuators, controlled variables and profiles. In AT discharges, the multiple parameter profiles which define the plasma state (safety factor, plasma rotation and pressure, etc...) are known to be strongly coupled, and a limited number of heating and current drive (HCD) actuators are available for plasma control. The system identification algorithms take advantage of the large ratio between the magnetic and thermal diffusion time scales and have been recently applied, in their full version, on both JT-60U and DIII-D data. On JT-60U, an existing series of high-bootstrap-current (70%), 0.9 MA non-inductive AT discharges were used. The actuators consisted of four groups of neutral beam injectors corresponding to on-axis perpendicular NBI, off-axis perpendicular NBI, on-axis co-current tangential NBI and off-axis co-current tangential NBI. On DIII-D, actuator modulation experiments were carried out in the loop voltage control mode to avoid feedback in the response data from the primary circuit. The plasma current varied between 0.7 and 1.2 MA. In addition to the loop voltage, the four HCD actuators were co-current, counter-current and balanced NBI, and electron cyclotron current drive. The reference plasma state was that of an AT scenario which had been optimized to combine non-inductive current fractions near unity with $3.5 < \beta_N < 3.9$, bootstrap current fractions of $> 65\%$, and $H_{98(y,2)} = 1.5$. The determination of the device-specific, control-oriented models that are needed to compute optimal controller matrices is discussed. The response of the relevant parameter profiles to variations of specific actuators can be satisfactorily identified from a small set of dedicated experiments. This provides, for control purposes, a readily available alternative to first-principle plasma modeling.

[1] D. Moreau, et al., Nucl. Fus. 48 (2008) 106001.

EXW/P2-08

Demonstration of 200 kA CHI Startup Current Coupling to Transformer Drive on NSTX

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Discharges started by Transient Coaxial Helicity Injection (CHI) in NSTX [1] have attained peak currents up to 300 kA for the first time. These discharges are coupled to induction, producing up to 200 kA additional current over inductive-only operation. For the first time, the CHI-produced toroidal current that couples to induction has continued to increase with the energy supplied by the CHI power supply. CHI in NSTX has shown to be energetically quite efficient, producing a plasma current of about 10 A/Joule of capacitor bank energy. These results indicate the potential for substantial current generation capability by CHI in NSTX and in future toroidal devices.

In earlier work, up to 50 kA of toroidal plasma current produced by the non-inductive method of CHI was successfully coupled to inductive ramp-up in NSTX. In those experiments, the CHI current that could be successfully coupled was limited by impurity production in the divertor region and the occurrence of absorber arcs (i.e. parasitic discharges across the insulating gap in the upper divertor). More recently, extensive conditioning of the divertor plates (CHI electrodes) greatly reduced impurity production during CHI. Further, by energizing, for the first time, the axisymmetric absorber field-nulling coils located near the upper divertor in NSTX, the absorber arcs could be delayed or suppressed. Also, the use of evaporated coatings of lithium on the plasma facing components in NSTX [2] increased the current at the hand-off from CHI to induction to nearly 200 kA. Later in the inductive ramp-up, the discharges with CHI applied reached significantly higher plasma current than discharges with only the inductive loop voltage applied.

These results represent a factor of four improvement in the magnitude of current that was ramped up by induction, a factor of three increase in the initial start-up current, and the first results demonstrating flux savings in NSTX. The CHI started discharge, when coupled to induction produces about 60% more current than the comparison inductive-only case. These results confirm that CHI could be an important tool for non-inductive start-up in next-step STs.

[1] R. Raman et al., Nuc. Fus., 49, (2009).

[2] H.W. Kugel et al., J. Nuc. Mat., 390-391 (2009).

EXW/P2-09

Observation and Analysis of a High Frequency MHD Activity during Sawteeth in KSTAR Tokamak

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During the 2009 campaign of KSTAR we tried to find a signature of internal kink/electron fishbone instability by using ECH off-axis heating and ECCD current drive at high field side of $q = 1$ surface. Our objectives were to identify, 1) the sawtooth oscillation, and 2) the radius of $q = 1$ surface from ECE signal data, and 3) MHD activities from Mirnov coil data. We found a high frequency $4 - 7$ kHz MHD mode driven by ECRH/ECCD during the sawteeth.

EXW/P2-10

Transient Process of A Spherical Tokamak Plasma Startup by Electron Cyclotron Waves

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The results of non-inductive startup in the SUNIST spherical tokamak (R/a: 0.3 m/ 0.23 m; B_{T0}: 0.15 T) by a 2.45 GHz microwave through electron cyclotron resonance heating (ECRH) are presented. Two discharge regimes with different transient characters, which determine the plasma current, are observed. The transient processes of discharges are experimentally investigated by scanning the radial resonance position, vertical field and the microwave power. Analysis of the measurements of microwave reflection and visible light emission prompt a process dominated by the combination of ionization, loss along the open field line and the gradient B drift. The discharges are modeled in one dimension. The simulation results qualitatively agree with the experiments. Both experiments and simulations suggest that the discharge character has less dependency on the experimental parameters except gas filling pressure confirming that the control of filling pressure is of great importance for startup a spherical tokamak by ECRH using low frequency microwaves.

EXW/P2-11

RF Plasma Production and Heating Below Ion-Cyclotron Frequencies in Uragan Torsatrons

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In IPP-Kharkiv there are two torsatrons (stellarators) in operation. A smaller one, Uragan-3 M, has $l = 3$, $m = 9$, $R = 1$ m major radius, $\langle a \rangle < 0.12$ m average plasma radius and toroidal magnetic field $B < 1$ T. The whole magnetic system is enclosed into a large 5 m diameter vacuum chamber. Uragan-2M is a medium-size machine with $l = 2$, $m = 4$ and reduced helical ripples. This torsatron has the major plasma radius $R = 1.7$ m, the average minor plasma radius $\langle a \rangle < 0.24$ m and the toroidal magnetic field $B < 1$ T. The Alfven resonance heating in a high key-parallel regime is used on both machines. This method of heating is advantageous for small size devices since the heating can be accomplished at lower plasma densities than the minority and second harmonic heating. Both machines equipped with two antennas. One is a frame-type antenna for low density plasma production. Another antenna in Uragan-3 M is an unshielded THT (three-half-turn) antenna [1] that consists of 3 straps oriented in poloidal direction. In regular discharges the frame antenna creates plasma with the density $\langle n \rangle \sim (0.5-2) \times 10^{12}$ cm^{-3} and electron temperature $\langle T \rangle \sim 1$ keV [2]. The THT antenna is usually not used to produce plasma and, therefore, its pulse follows the pulse of the frame antenna. A series of experiments is performed aimed to study the features of the discharge with the THT antenna. Electron temperatures in the range $\langle aT_e \rangle \sim 0.2 - 0.4$ keV are achieved at the plasma densities an order of magnitude higher than produced by the frame antenna $\langle n \rangle \sim (0.5 - 1.5) \times 10^{13}$ cm^{-3}. Plasma energy content is increased up to 5 times. A new 4-strap shielded antenna is manufactured and installed in Uragan-2M [3]. First experimental data for radio-frequency heating with this antenna are presented.

[1] V.E. Moiseenko in IAEA Technical Committee Meeting (Proc. 8th Int. Workshop on Stellarators, Kharkov 1991), IAEA, Vienna, 207 (1991).

[2] O.M. Shvets, I.A. Dikij, S.S. Kalinichenko et al., Nucl. Fusion 26, 23 (1986).

[3] V.E. Moiseenko, Ye.D. Volkov, V.I. Tereshin, Yu.S. Stadnik, Plasma Physics Reports 35, 828 (2009).

EXW/P2-12

Generation of Initial Closed Flux Surface by ECH at Conventional Aspect Ratio of R/a $\sim$ 3; Experiments on the LATE device and JT-60U Tokamak

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Non-inductive startup of tokamaks is a key issue for an economical reactor and ECH is an attractive tool for the startup. Injection of an EC power to a toroidal field configuration with a weak vertical field (Bv) leads to a plasma current initiation, where the charge separation caused by the grad-B drifts is shorted by parallel currents along helical B and generates a plasma current. The ratio of the self-poloidal field by this current to Bv is inversely proportional to the aspect ratio, suggesting that the closed flux surface formation becomes difficult with the increase of aspect ratio. While an initial closed flux surface was shown to be generated by ECH in a number of low aspect ratio devices, there was no report at the conventional aspect ratio except for one in DIII-D. In this paper we report experiments and analyses on LATE and JT-60U, showing that the closed flux surface formation by ECH is still possible in $R/a \sim 3$ with an appropriate Bv configuration.

To conduct experiments at conventional device aspect ratios of $R(0)/a = 2 - 3$ in LATE, a movable inboard limiter and an outboard limiter have been installed. After a microwave power of 2.45 GHz 30 kW is injected to the device with $R(0)/a = 3$, a plasma current is initiated and increases slowly up to 0.5 kA, and then jumps up to 0.9 kA (current jump). Magnetic analysis and visible camera images show that the plasma current starts to flow on the outboard side apart from the ECR layer. As the current increases the current profile is broadened and shifts to the inward, and touches the ECR layer just before the current jump. Then the plasma current spontaneously jumps up and closed flux surfaces are formed.

The spontaneous formation depends significantly on the Bv decay index. As the aspect ratio increases, higher decay indices are required for the current jump. This suggests that mirror confinement of the EC heated electrons is essential for the current jump.

Similar current startup discharges have been preformed on JT-60U ($R/a \sim 3$). Up to 4 MW of 110 GHz, O-mode ECH was applied. We set the Bv decay index to be within the current jump condition in LATE. A plasma current is initiated and increased to 20 kA. Magnetic analysis shows that the current is initiated on the outboard and shifts to the inward, and finally the current profile touches the ECR layer, which are similar to the profiles in LATE.

EXW/P7 — Poster Session

EXW/P7-01

Radiofrequency Current Drive Experiments on MST

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Current profile control is a crucial tool for understanding the drive of tearing fluctuations in the reversed field pinch. Simulations of auxiliary edge parallel current drive predict a reduction of tearing activity, and indeed in experiment there is a significant decrease of magnetic fluctuations with inductive edge current drive in the MST. This in turn leads to a dramatically increased (factor of 10) electron energy confinement time and evidence that transport is no longer dominated by magnetic turbulence. The use of rf waves to drive edge current offers steady and more precise profile control than the existing inductive approach, which is transient, radially diffuse and induces a large change to the magnetic equilibrium. Lower Hybrid (LH) and electron Bernstein waves (EBW) are being studied as candidates for the overdense, high beta plasma. Ray tracing and Fokker-Planck calculations predict good absorption and directional control for both waves, as required for effective current drive. The lower hybrid studies involve novel antenna design and extending LH physics to plasmas with high dielectric constant. In contrast, the EBW studies benefit from simpler antenna requirements, but the mode conversion wave physics needs to be established for a high beta RFP plasma. At present, localized x-ray emission has been observed with rf injection at the 100 kW level of each wave. Toroidally localized hard x-ray (HXR) emission with energy as high as 50 keV is observed during LH injection (with $\sim$ 125 kW coupled to the plasma). The flux and energy spectrum is consistent with acceleration of plasma electrons in the antenna near field with electric fields computed by electromagnetic modeling. Enhanced SXR emission ($4 - 7$ keV) is observed during EBW injection ($\sim$ 100 kW of 3.6 GHz launched) when accompanied by a period of low magnetic fluctuations. Hardware for the EBW current drive project is being upgraded to use a 1 MW, 5.5 GHz klystron, with the higher frequency (shorter wavelength) enabling the use of a smaller port hole for the launching antenna.

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EXW/P7-02

Lower Hybrid Current Drive at Densities Required for Thermonuclear Reactors

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For the progress of the thermonuclear fusion energy research based on the tokamak concept, to drive current non-inductively in high-density plasmas is essential for producing steady-state, thermally well insulated, stable and large volume plasmas. ITER will have relatively high densities at the periphery of the plasma column ($n_{e-0.8}$ about $(0.8-1)\times 10^{20}$ m^{-3} at normalised minor radius r/a about 0.8). The lower hybrid current drive (LHCD) effect, produced by externally launched LH waves, is potentially the most suitable tool for driving current at large radii of fusion relevant plasmas, consistent with the needs of ITER. However, previous experiments showed that plasma-wave interaction signatures decrease with density much more than expected by standard theory. Such experimental behaviour should be possibly interpreted as the effect of the spectral broadening, produced by non-linear wave-wave interactions at the plasma edge. When operating in LHCD experiments at relatively high plasma edge densities, the strong parametric instability (PI)-produced spectral broadening prevents penetration in the plasma core. By numerical investigation, it was also predicted that the PI effect is mitigated when operating with relatively high electron temperatures at the plasma periphery, resulting in allowing wave penetration into the core of high-density plasmas. In order to test the effect of higher edge temperature ($T_{e-outer}$) in producing condition of LH power penetration at high densities. In this high $T_{e-outer}$ regime supra-thermal electrons produced by the coupled LH power in the core (at r/a about $0.3-0.4$) occur even at the highest performed plasma densities: $\langle n\rangle$ about 2×10^{20} m^{-3}. In this condition, we have: $n_{e0}=(4.2\pm 0.5)\times 10^{20}$ m^{-3}, $n_{e-0.8}=(0.84\pm 0.1)\times 10^{20}$ m^{-3}, and the FEB emission radial profile peaks at r/a of about $0.3-0.4$. The experiment confirms the theoretical prediction that identifies operations at relatively high temperature at the plasma periphery as the key for reducing the parametric instability-induced spectral broadening and allowing penetration of lower hybrid waves into high-density plasma. Thus, the how is now available for extending the range of usefulness of the lower hybrid current drive effect to regimes of critical importance for fusion reactors.

EXW/P7-03

Recent Experiments of Lower Hybrid Wave-Plasma Coupling and Current Drive in EAST Tokamak

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A 2 MW 2.45 GHz lower hybrid wave (LHW) system has been installed and run in EAST tokamak, in which LHW coupling and lower hybrid current drive (LHCD) experiments have been performed in both divertor configuration and limiter plasma with parameters of a plasma current of 250 kA and a central line averaged density of $1.0 - 1.3 \times 10^{19}$ m^{-3}, suggesting that present LHW system is effective to couple LHW into plasma and drive plasma current with low reflection coefficient (RC) of $\sim 10\%$. Results show that RC firstly decreases with increasing distance between plasma and LHW grill, and then increases with the distance, in agreement with the simulation through Brambilla theory. Studies indicate that with the same plasma parameters, the best coupling is obtained in the limiter case, followed by the single null, and the last one is the double null configuration. The main reason for this is the different magnetic connection length. Current drive efficiencies investigated by a least squares fit show that there is no obvious difference in the drive efficiency ($\sim 0.73 \times 10^{19}$ Am^{-2}W^{-1}) between the double null and the single null cases, whereas the efficiency is a little small in the limiter configuration (0.58×10^{19} Am^{-2}W^{-1}), in agreement with the ray tracing code simulation by LUKE/C3PO. The different current efficiency can be explained by the power spectrum up-shift factor estimated from a power flow process in different plasma configurations. There is little dependence of drive efficiency on LHW power spectrum. The possible reason is that the temperature is low and LHW is absorbed after multi-pass propagation.

EXW/P7-04

Latest Achievements of the JET ICRF Systems in View of ITER

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The capability of Ion Cyclotron Resonance Frequency (ICRF) systems to reliably inject high power levels into ELMy plasmas is essential both for the JET research program and for future ITER operations. In parallel with the installation and commissioning of the ITER-like antenna (ILA), the conventional ICRF A2 antenna system has been significantly modified to improve its performance.

The ILA consists of a closely packed array of 4 Resonant Double Loops with in-vessel matching capacitors and low impedance Conjugate-T points. It has demonstrated efficient trip-free ELM tolerant operation on ELMy plasmas with high power densities (up to 6.2 MW/m^2 on L-mode and 4.1 MW/m^2 on H-mode) and voltages up to 42 kV without significant impurity production. The ILA further provided validation of the ICRF coupling modeling code Topica. This is relevant for antenna design for ITER as it demonstrates that with ITER's presently estimated density profiles its proposed ICRF system is able to couple 20 MW with a system voltage below 45 kV.

The conjugate-T junction principle was also used by pairing straps from A2 antennas C and D thus creating 4 conjugate-T points outside the vacuum vessel. Additionally, the load tolerant circuit installed in 2005, based on 3 dB hybrid couplers feeding straps from A2 antenna arrays A and B in pairs, was updated for more extensive use. Both load tolerant approaches have demonstrated robust and reliable performance during operations in a variety of plasma loading conditions including on large Type I ELMs. Depending on the plasma scenario, the trip-free time-average power levels coupled to ELMy H-mode plasma during long pulse operations have reached 3.3 MW for the 3 dB hybrid and 4 MW for the External Conjugate-T (ECT) system. Up to 7 MW total ICRF power was coupled to ELMy plasmas simultaneously by all four conventional JET ICRF antennas.

As for the ILA, successful implementation of ELM-tolerant systems requires to successfully distinguish arcs from ELMs. The developed Advanced Wave Amplitude Comparison System (AWACS) has proven highly effective for arc protection of the ECT system, while the existing compromise between the Voltage Standing Wave Ratio requirements for the ELM-tolerance and arc protection of the 3 dB hybrids still requires resolution by introduction of new arc detection techniques.

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EXW/P7-05

First Experimental Results with the ITER-Relevant Lower Hybrid Current Drive Launcher in Tore Supra

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The new ITER-relevant Lower Hybrid Current Drive (LHCD) launcher, based on the Passive Active Multi-junction (PAM) concept, was brought into operation on the Tore Supra tokamak in October 2009. The PAM launcher concept was designed in view of ITER to allow efficient cooling of the waveguides due to plasma radiation, RF losses and neutron damping. In addition, it offers low power reflection close to the cut-off density, which is important in view of ITER, where the large distance between the plasma and the wall may bring the density in front of the launcher to low values. The complete system operated successfully on Tore Supra already on the second experimental day, coupling 0.4 MW into the plasma for 4.5 s. The maximum power and energy reached so far, after ~ 400 pulses on plasma, is 2.7 MW during 77 s (exceeding 200 MJ injected energy). This corresponds to a power density of 25 MW/m^2, i.e. its design value at $f = 3.7$ GHz. The main goals of the initial experimental campaign were to: i) compare the coupling behaviour of the PAM launcher to the predictions from the ALOHA code, ii) demonstrate reliable power handling during edge perturbations mimicking ELMs and iii) achieve ITER-relevant power densities in pulse lengths exceeding several seconds.

The power reflection coefficient (RC) behaviour on the PAM launcher shows good agreement with calculations, carried out with the ALOHA code. The average RC is $< 2\%$ when the electron density in front of the launcher is close to the cut-off density. When the density increases, RC increases, as predicted by modelling. Coupling experiments with edge perturbations, produced by supersonic molecular beam injection (SMBI) to simulate ELM-behaviour, were carried out at intermediate LHCD power ($\sim$ 1.5 MW). During a SMBI, the electron density in front of the PAM launcher increases typically by a factor of four and the RC increases, in accordance with modelling. The applied power remained constant during the SMBI, demonstrating an ELM-resilient system. In standard coupling conditions, the maximum power and energy achieved on the PAM launcher so far is 2.7 MW during 77 s, corresponding to a power density of 25 MW/m^2. In addition, 2.7 MW was coupled at a plasma-launcher distance of 10 cm. The launcher front face security, based on infrared imaging and CuXIX-line emission, showed no arcing at the PAM launcher mouth during high LHCD power.

EXW/P7-06

Observation of Global Alfvèn Eigenmode Avalanche-like Events on the National Spherical Torus Experiment

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This paper presents the first observations of Global Alfvén Eigenmode (GAE) avalanches and concomitant fast ion redistribution. Super-Alfvénic ion populations, like the fusion-alpha's on ITER, can excite instabilities extending from low frequency Energetic Particle Modes (EPMs), through Toroidal Alfvén Eigenmodes [TAE] to Global and Compressional Alfvén Eigenmodes [GAE and CAE] in the frequency range of roughly $0.1\omega_{ci}$ to $0.7\omega_{ci}$. The GAE instabilities on NSTX exhibit complex non-linear behavior, including strong growth which onsets above an amplitude threshold, when resonance regions in phase space start to overlap, resulting in enhanced rapid growth and fast ion redistribution. No neutron drops or other direct indications of fast ion loss or redistribution are correlated with the strong GAE avalanche bursts, however, the modes are suppressed following the avalanche, suggesting depletion of fast ion density in the velocity space driving the modes and in some instances the GAE bursts appear to trigger lower frequency energetic particle driven activity, either TAE avalanches or Energetic Particle Modes, suggesting some significant redistribution of fast ions in phase space has occurred. This paper also provides some of the first measurements of internal GAE mode structure showing the mode amplitude peaks towards the plasma core.

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EXW/P7-07

Fast-Ion Transport Induced by Alfvén Eigenmodes in ASDEX Upgrade

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On the road to ITER, a basic understanding of the interplay between fast-particles and Alfven Eigenmodes (AEs) is mandatory. In ASDEX Upgrade, an extended suite of diagnostics now allows detailed measurements of AE spatial structure and the subsequent fast-ion transport. RSAEs and TAEs have been driven unstable by fast-ions from ICRH as well as NBI origin. In ICRF heated plasmas, diffusive and convective fast-ion losses induced by AEs have been characterized in fast-ion phase space. While single RSAEs and TAEs eject resonant fast-ions in a convective process directly proportional to the fluctuation amplitude, δB, the overlapping of multiple RSAE and TAE spatial structures and wave-particle resonances leads to a large diffusive loss, scaling as δB^2. In beam heated discharges, coherent fast-ion losses have been observed primarily due to TAEs. Core localized, low amplitude NBI driven RSAEs have not been observed to cause significant coherent fast-ion losses. The fast-ion redistribution due to RSAEs/TAEs has been studied using Fast-Ion D_α (FIDA) spectroscopy and will be presented. The AE radial and poloidal structures have been obtained with unprecedented details using a fast 1D and 2D ECE radiometer.

EXW/P7-08

Investigation of Beams and Waves Plasma Interaction in the Globus-M Spherical Tokamak

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Last two years research results on beams - plasma and waves - plasma interaction are reported. The Globus-M spherical tokamak has an unique possibility to investigate effects produced on plasma via injection of a low energy but high density beam (produced by a plasma gun) as well as high energy, low density beam (produced by NB injector). The characteristic beam energies differ by two orders of magnitude (0.3 keV and 30 keV respectively). Two orders of magnitude ratio of frequencies is typical for the RF methods of heating and CD applied to Globus-M plasma. ICRH frequency ($7.5-9$ MHz) is about 100 times lower than LHCD frequency (0.9 GHz) utilized in first experiments. Plasma fueling, heating and CD experiments performed on Globus-M with the help of launching of beams of different energies and waves of different frequencies are discussed. Underlying physical processes are outlined and analyzed.

EXW/P7-09

Characteristics of Anomalous Transport and Losses of Energetic Ions caused by Alfvénic Modes in LHD Plasmas

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Interplay between energetic ions and energetic-ion driven MHD instabilities is one of the key physics issues in current fusion experiments since Alfvénic modes such as toroidal Alfvén eigenmodes (TAE) and the energetic-particle modes may lead to anomalous loss of energetic alpha particles in a future D-T burning plasma. In this work, anomalous transport and losses of energetic ions caused by Alfvénic modes in neutral beam (NB)-heated Large Helical Device (LHD) plasmas are studied with a variety of fast-particle diagnostics. Correlated with recurrent bursting TAEs, significant increases of energetic-ion losses are observed. It is revealed that co-going transit energetic ions are responsible for destabilization of TAE in LHD and energetic ions are anomalously expelled through a diffusive-type loss mechanism. Also, experimentally observed TAEs can be explained in both frequency and radial position using a computational MHD analysis. Using the internal structure of TAE predicted by AE3D, efforts on beam-ion simulation with perturbed magnetic field are made to verify whether the experimental observation for beam ions losses can be explained by the model calculation. The result indicates that the perturbed field can increase beam-ion losses compared with that for the stationary field. Also, the pitch angle and energy of escaping beam ions predicted by the particle simulation matches approximately those of experimental observations.

EXW/P7-10

Fast Ion Effects during Test Blanket Module Simulation Experiments in DIII D

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The confinement of fast beam-ions was studied in DIII-D in the presence of a scaled Mock-up of two Test Blanket Modules (TBM) for ITER. The TBM on DIII-D has four protective carbon tiles arranged vertically with a thermocouple placed in the center of the back of each tile. Heat loads on the four plasma facing carbon tiles were measured and temperature increases of up to 200°C were found for the two central tiles closest to the midplane when the TBM fields were activated. The temperature increase of the two outer tiles was found to be an order of magnitude lower than the two central tiles. When the TBM fields were off a temperature rise of 10°C or less was observed for all four tiles in identical discharges. These measurements agree qualitatively with results from the full orbit-following beam-ion code, SPIRAL, that predict beam-ion losses localized to the central two carbon tiles near the midplane when the TBM fields are energized.

Fast-ion loss diagnostics, such as fast-ion D_α (FIDA) and neutron scintillators, were used to detect possible signs of central fast-ion loss or redistribution. Within the experimental uncertainties no significant change in the fast-ion population was found in the core of these plasmas. These results are consistent with SPIRAL analysis.

From these experiments it can be concluded that the TBM fields do not affect the fast-ion confinement in a harmful way, which is good news for ITER.

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EXW/P7-11

First Results from ICRF Heating Experiment in KSTAR

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KSTAR is the Korean national superconducting tokamak aiming at high beta operation based on AT scenarios, and ICRF is one of the essential tools to achieve this goal. The Faraday shield and current straps have water-cooling channels built in for the eventual long pulse operation of KSTAR. Operation of the transmitter with an output power of 2 MW for 300 s over the operation frequency range of 30−60 MHz, involves very severe conditions regarding stable operation of the system. After several failures of the amplifiers during the factory and site acceptance test, we eventually achieved a continuous output power of more than 1.9 MW for 300 s over the whole operation frequency range. Commissioning of the ICRF system including the antenna was carried out along with the first plasma experiments of KSTAR in 2008 and about 150 kW ICRF power was successfully injected to the circular limited plasma with resonant double loop antenna, which may accumulate ITER-relevant operational experiences at the superconducting tokamak devices. In 2009 experimental campaigns, up to 300 kW has been successfully applied and about 50% increase of electron temperature was observed with RF power injection in the Fast Wave Electron Heating(FWEH) mode. According to the spectroscopic and residual gas analyses in 2009, H/D ratio and hydrogen retention in the wall is very high. Thus wall conditioning including high temperature baking on PFC is necessary for efficient minority ion heating experiments planned for 2010. H minority heating in a diverted plasma configuration will soon be carried out in 2010, and the results will be presented in the conference.

EXW/P7-12

Recent Developments in High-Harmonic Fast Wave Physics in NSTX

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Understanding the interaction between ion cyclotron range of frequency (ICRF) fast waves and the fast-ions created by neutral beam injection (NBI) is critical for future devices such as ITER, which rely on a combination ICRF and NBI. Experiments in NSTX which use 30 MHz High-Harmonic Fast-Wave (HHFW) ICRF and NBI heating show a strong competition between electron heating via Landau damping and transit-time magnetic

pumping, and rf acceleration of NBI generated fast ions. Understanding and mitigating some of the rf power loss mechanisms outside the last closed flux surface (LCFS) has resulted in improved rf heating inside the LCFS. First wall lithium coating was found an effective tool in achieving these goals. A similar magnitude of rf power is absorbed by the fast ions and electrons within the LCFS.

Experimental observations point toward the rf excitation of surface waves, which disperse wave power outside the LCFS, as a leading loss mechanism. Lithium coatings lower the density at the antenna, thereby moving the critical density for perpendicular fast-wave propagation away from the antenna and surrounding material surfaces. Visible and infrared imaging reveal flows of rf power along open field lines into the divertor region.

When rf power is coupled to NBI plasmas the neutron production rate (Sn) increases and the fast-ion profile broadens and is enhanced, indicative of a significant interaction of fast waves with NBI fast ions. Power deposition into the fast ions can be estimated by comparing the measured Sn to that predicted by the TRANSP code which includes the TORIC full-wave code, and by the CQL3D Fokker-Planck code.

EXW/P7-13

Neutron Flux Measurements in ICRF Plasmas on HT-7 and EAST

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Ion cyclotron range of frequencies (ICRF) heating combined with lower hybrid current drive (LHCD) experiments have been carried out on HT-7. ICRF experiments have been carried out at three frequencies, which correspond to different heating scenarios. It was observed the fusion neutron flux only increases remarkably at frequency of 24 MHz in mode conversion (MC) scheme with the ion-ion hybrid resonant layer near the centre of plasma. The neutron flux is 2.1 powers of ICRF power. Since there is no direct ion temperature measurement in HT-7, the main ion temperature is increased by a factor of 9 at 0.3 MW assuming all neutrons from thermal D-D reaction. However, storage energy measurement does not support this statement. There are several possibilities to account this anomalous increment of neutron flux: (a) high harmonic cyclotron resonance of light impurities could be excluded since there are no cyclotron resonant layers near the centre; (b) Ion Bernstein wave (IBW) or ion cyclotron wave (ICW) heating ions by cyclotron resonance. There is a possibility of the IBW heating by nonlinear 3/2 harmonic deuterium resonance located near the plasma centre; (c) possible contribution from high energy tail on the ion energy distribution function produced via second harmonic deuterium cyclotron resonant layer located at 140 cm of low magnetic field side (LFS). EAST is equipped with higher ICRF power, and diagnostics will provide direct ion temperature measurements. This provides possibility to investigate anomalous increase of neutron flux by excluding contribution from direct ion heating in ICRF MC scenarios.

EXW/P7-14

Nonlinear Evolution of Beam Driven Waves on MAST

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Experiments on Alfvénic instabilities driven by super-Alfvénic beams in the spherical tokamak MAST have exhibited a variety of modes excited in a broad range of frequencies from Alfvén Cascade eigenmodes, Toroidal Alfvén Eigenmodes, and chirping modes in the frequency range $50-150$ kHz, to compressional Alfvén eigenmodes in the frequency range $0.4-3.8$ MHz, which is approaching the cyclotron frequency of plasma ions, $\omega \approx (0.1 \div 1)\omega_{Bi}$. For energetic ions produced via neutral beam injection, the unstable distribution function is formed by Coulomb collisions, with dynamical friction (drag) and velocity space diffusion dominating in different regions of phase space (separated by the critical velocity $V_{\rm crit}$).

The aim of the present work is to demonstrate that the nonlinear evolution of these modes is determined by the type and strength of relaxation processes. In particular, we validate the recent theoretical finding that drag encourages the beam-driven waves near marginal stability to follow an explosive scenario. An efficient numerical tool has now been created for this task, the results of which show similarities with features that are observed in recent and past experiments, indicating the central role that the drag might play in the evolution of beam or alpha particle driven waves in tokamaks. This has galvanised the current ongoing effort to perform quantitative modelling of Alfvénic instabilities in the presence of drag using the HAGIS code. The universal feature of these strongly nonlinear scenarios that incorporate drag is the asymmetry of the mode evolution with respect to the wave-particle resonance. As a result we observe a transition from a steady state non-linear wave to one with a hooked frequency chirp.

To connect the bump-on-tail model to the experiments we characterise the relative importance of the drag and the velocity space diffusion by considering the wave particle resonance condition, $\Omega = k_{\|}V_{\|\rm res} - \omega + p\omega_{Bb} - \mathbf{kV_{Db}} = 0$, and calculate the width of the resonance due to diffusion $\Delta\Omega_{diff}$ and drag $\Delta\Omega_{drag}$ from the the Fokker-Planck operator for different types of modes. The importance of the parameter $V_{\|\rm res}/V_{\rm crit}$ is assessed.

EXW/P7-15

Studies on Neutral Beam Ion Confinement and MHD Induced Fast-Ion Loss on HL-2A Tokamak

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Experiments with a high-energy deuterium neutral beam (NB) injection (30 keV, about 0.6 MW) were performed on the HL-2A tokamak. The emission of d-d fusion neutrons dominated by beam-plasmas reactions when the deuterium NB was injected into the deuterium target plasma was observed by means of a 235 U fission chamber. To obtain information on NB deposition and the slowing down of beam ions in HL-2A plasmas, a very short-pulse deuterium NB injection, or the so-called “blip” injection, was applied into MHD-quiescent Ohmic deuterium plasmas. Analysis of neutron decay following the NB “blip” injection indicates that tangentially injected beam ions are well confined, slowing down classically in the HL-2A.

In contrast to the MHD-quiescent plasma, anomalous losses of beam ions were observed when a core localized mode with a frequency up-chirping from 15 to 40 kHz and an $m = 2/n = 1$ tearing mode instability were concurrently present in the plasma. The core localized mode was identified as a beta-induced Alfvèn acoustic (BAAE) mode by its frequency up-chirping behavior and numerical calculation with the MEGA code. Such concurrent multiple modes led to fast-ion loss coupling, showing a strong influence of a core-localized fast-ion driven BAAE mode together with the tearing instability on the fast-ion transport. Furthermore, a clear splitting was firstly observed for such an Alfvén-acoustic type mode. The frequency splitting is found to be strongly linked to the effect of the Doppler shift due to NBI-induced plasma rotation, providing further insights into how frequency splitting structures are generated in plasma.

EXW/P7-16

Investigation of Fast Pitch Angle Scattering of Runaway Electrons in the EAST Tokamak

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An experimental study of the investigation of fast pitch angle scattering (FPAS) of runaway electrons in the EAST tokamak has been performed. From the newly developed infrared detector (HgCdTe) diagnostic system, the infrared synchrotron radiation emitted by relativistic electrons can be obtained as a function of time. The FPAS is analyzed by means of infrared detector diagnostic system and the others correlative diagnostic systems (including electron cyclotron emission, hard x-ray, neutrons). When the velocity of

runaway electrons reaches the critical value, the Parail and Pogutse instability is excited. Thereafter the parallel velocity is transformed into perpendicular velocity because of the interaction between energetic runaway electrons and lower hybrid waves are excited by Cerenkov resonance of the electrons on Langmuir waves. Anomalous Doppler resonance will result in pitch angle scattering of the runaway electrons if the resonance criterion is fulfilled. Therefore, the pitch angle of high energetic runaway electrons increases immediately (FPAS). The intensity of infrared synchrotron radiation and the ECE signal increase rapidly at the time of FPAS because of the fast increase of pitch angle and the perpendicular velocity of the runaway electrons. Neither the two stream instability nor the excitation of parameter instability or the instability evocated by a positive slope in the distribution function are considered likely to explain the observed phenomena. since the bulk plasma is nearly unaffected by the instability; the FPAS is independent of the number of high energy electrons and the step-like increase of the ECE signal while the absence of an oscillating character of the infrared synchrotron radiation. It is unlikely that the magnetic field ripple resonance causes the FPAS analyzed above. Therefore, the Parail and Pogutse instability is a possible mechanism for the FPAS.

EXW/P7-17

Alfven Eigenmodes Properties and Dynamics in the TJ-II Stellarator

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Energetic ion driven Alfven Eigenmodes (AEs) in the NBI-heated plasma at the TJ-II stellarator were studied by Heavy Ion Beam Probing (HIBP) in the core, and by Langmuir and Mirnov probes (MP) at the edge. HIBP observed the locally (~ 1 cm) resolved AE at radii $-0.8 < \rho < 0.9$. The set of AE branches with low poloidal numbers ($m < 8$) was detected by MP. AEs on the density, electric potential and poloidal magnetic field oscillations were detected by HIBP at frequencies 50 kHz $< \omega(AE) <$ 300 kHz with a high resolution ($<$ 5 kHz). The MP and HIBP data have a high coherency for specific branches of AE. When the density rises, AE frequency is decreasing, but the cross-phase between the density and potential oscillations remains permanent. Comparison with computational MHD mode predictions indicates that some of the more prominent frequency branches can be identified with radially extended HAE (helical), GAE (global) and TAE (toroidal) modes. Example of such identification for observed AE with frequency 177 kHz is given. The mode with couplings between m, $n = (1, -1)$ and $(3, -5)$ components was classified as HAE. The turbulent AE driven flux was small in comparison with the broadband turbulent flux for the same frequency domain.

EXW/P7-18

Comparison of Central Fast Ion Distributions between Plasmas with on-Axis and off-Axis NBI Current Drive on ASDEX Upgrade

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In addition to providing a non-inductive current drive source, current drive from neutral beam injection (NBCD) can play a key role in advanced tokamak scenarios where where off-axis current drive flattens the q-profile leading to enhanced confinement regimes. However, experiments of off-axis NBCD have unveiled deviations from the classical picture under certain operating regimes even in the absence of MHD instabilities [1-3]. This led to studies of the understanding of turbulence driven fast ion transport on time scales smaller than the current redistribution time [4-7] specifically at larger plasma radius where the plasma fluctuations are larger. This paper reports comparison of the near central fast ion distribution with on-axis and off-axis NBCD configuration in low triangularity discharges. The fast ion distribution is measured by the microwave based collective Thomson scattering (CTS) diagnostic installed on ASDEX Upgrade [8]. The CTS diagnostic on AUG is capable of measuring localized 1D ion velocity distributions and anisotropies dependent on the angle of the fluctuation wave vector (k^δ) to the magnetic field defined by the scattering geometry. The 1D velocity resolved distribution function $g(u)$ is the projection of the 2D ion velocity distribution onto k^δ. The results show marginal difference in the fast ion distribution function, within the error-bars, between the two NBI heating configurations suggesting an enhanced fast ion diffusion. Both discharges are MHD quiescent according to magnetic and soft X-ray measurements. Comparison of the fast ion distribution function between CTS measurements and TRANSP/NUBEAM simulation (without additional fast particle diffusion) show a much better agreement for the NBCD on-axis case. This suggests enhanced fast ion diffusion for the off-axis heating configuration. However, within the error bars, the results do not show strong evidence of energy dependent transport.

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EXW/P7-19

Experimental Study of Second Harmonic ECCD in Heliotron J

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Second harmonic electron cyclotron current drive (ECCD) experiments have been made in Heliotron J by using an upgraded EC launching system. A focused Gaussian beam is injected with the parallel refractive index, $N_{\parallel}$, ranging from -0.05 to 0.6. According to ray tracing calculations, the focused Gaussian beam with controllable injection angle makes the EC power absorption localized. The experimental results show that the EC driven current can be controlled by $N_{\parallel}$, and it depends on the local magnetic field structure where the EC power is deposited. The EC current is more driven when the EC power is deposited at the position where the Ohkawa effect is weak. A large increase in ECE signals has been observed when the EC current was driven, indicating the important role of high-energy electrons on the ECCD.

EXW/P7-20

Dynamics of Fast Ions during Sawtooth Oscillations in the TEXTOR Tokamak measured by Collective Thomson Scattering

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Fast ions generated by fusion reactions and by auxiliary heating play an important role in tokamak plasmas. The interaction between plasma instabilities and fast ions needs to be understood in order to optimize burning plasmas. Collective Thomson scattering (CTS) provides measurements of 1D projections of the fast-ion velocity distributions. In Ref. [1] first fast-ion CTS measurements in neutral beam injection (NBI) heated TEXTOR plasmas with sawtooth oscillations were reported. The measured 1D fast-ion distribution was found to drop up to 50% for resolved directions with a significant component parallel to the magnetic field due to a sawtooth crash. Furthermore, measurements in which the velocity distribution was resolved close to perpendicular to the magnetic field revealed no significant drop at the time of the sawtooth crashes. Here we expand on the discussion in Ref. [1] and include results for plasmas with NBI heating and additionally central ion cyclotron heating.

[1] S.K. Nielsen et al., submitted to Phys. Rev. Lett (2010).

EXW/P7-21

Advances in ICRF Physics and Technology on ASDEX Upgrade

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ASDEX Upgrade and its accompanying ICRF test stands are particularly well suited to advance ICRF towards ITER and beyond. The ability to operate ICRF on plasmas with strong ELMs has identified shortcomings in the arc detection methods, some of which have difficulties to discriminate between ELMs and arcs. The design and optimization of arc detectors require a precise understanding of the RF arc evolution in tuned transmission lines: the characteristics of such arcs are studied on ASDEX Upgrade and on a test bench to measure their noise, their light intensity and their reflected power. The research on arc detection and the development of arc detectors for ITER ICRF system has also lead to the investigation of plasma emission in the MHz range. By studying the noise generated by arcs in MHz range on ASDEX Upgrade, we discovered that the plasma also emitted three types of RF signals with frequencies in this range.

Operation with a metallic wall has made ICRF operation in ASDEX Upgrade, in particular with W coated limiters more difficult. Parallel electric antenna near-fields Epar are thought to cause high sheath voltages and sputtering at antenna limiters. Guidelines to reduce these fields, in particular those near the limiters, have been elaborated by near-field calculations with the HFSS code. A modification of existing ASDEX Upgrade ICRF antenna has been designed to test the approach based on the calculations before making more fundamental changes. The modification includes an alteration of the limiters to make them toroidally broader, a change of antenna straps by bias-cutting them such that the plasma-facing strap region is toroidally narrower, and an additional internal shield which carries a large fraction of image currents.

The antennas in ASDEX Upgrade can be connected such that two antennas operated simultaneously are situated on the opposite sides of torus and not connected via magnetic field lines. This allows a better characterization of the effect of RF power on the increased potentials on the field lines connected to the antenna and on W sputtering. Analysis of the experimental data is ongoing. It includes measurements by Langmuir probe techniques, W spectroscopy data and measurements of currents flowing via the antenna limiters.

EXW/P7-22

Evaluation of Fast-Ion Confinement with Three Dimensional Magnetic Field Configurations on the Large Helical Device

Osakabe, M.[1], Ito, T.[2], Murakami, S.[3], Yoshinuma, M.[1], Ida, K.[1], Kobayashi, M.[1], Bustos, A.[4], Castejón, F.[4], Isobe, M.[1], Kobayashi, S.[5], Ogawa, K.[6], Toi, K.[1], Takeiri, Y.[1], and LHD Experiment Group[1]

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Confinement of fast-ions is one of the most important issues in the magnetically confined fusion devices. Recently, it is recognized that non-axis symmetric ripple components of magnetic fields might affect the confinement properties of fast-ions in tokamak devices since the installations of TBM and RMP-coils break the axisymmetry of the configuration in ITER. The three dimensional (3D) ripple effect on fast-ion confinement is a common problem in helical devices. Since the 3D component of magnetic field can be changed more drastically in helical devices than in tokamaks, the 3D-effects on fast-ion confinement properties can be more clearly investigated in helical devices. On LHD, these effects were studied for various magnetic configurations by comparing the experimental results of the Fast Ion Charge eXchange Spectroscopy (FICXS) to those from numerical simulations. The result of this study will provide better understanding of alpha particle confinement in tokamak devices with various perturbation fields as well as in helical devices.

EXW/P7-23

Non-linear Dynamics of Toroidicity-induced Alfven Eigenmodes on NSTX

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The dynamics resulting from the non-linear coupling of multiple toroidicity-induced Alfvén eigenmodes (TAEs) is believed to be one of the main loss mechanisms for fast ions in ITER. This phenomenon is commonly observed in neutral beam-heated plasmas on the National Spherical Torus Experiment (NSTX), where bursts of TAE activity can cause substantial (up to 30%) losses over $\sim$ 1 ms. Fast ion losses scale with the activity in the TAE band. In addition, modes with frequencies both below and above the TAE gap appear in the Fourier spectrum of magnetic fluctuations during large amplitude bursts. The frequency and amplitude evolution of these modes is consistent with a simple model based on quadratic interactions between the unstable TAEs. This non-linear coupling leads to effective growth rates $>$ 10%, thus enabling an explosive behavior of TAE modes and enhanced fast ion losses.

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EXW/P7-24

Investigation of Runaway Electron Beam in EAST

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It is the first time to observe the synchrotron radiation emitted by runaway electrons with a visible CMOS camera in tokamak. Some information about runaway electron has been obtained in the EAST tokamak, which is the first full superconducting tokamak with non-circular cross-section in the world. Both the energy and pitch angle of runaway electrons have been derived from the measurement of synchrotron radiation. The synchrotron radiation of runaway electrons was observed in the center region in most cases. The normal synchrotron radiation pattern is ellipse with an inclination angle to the equator, which is consistent with the numerical calculation of runaway electron synchrotron radiation. In few cases, a stable shell shape of synchrotron radiation can be observed. The mechanism for the formation of the shell is unclear.

EXW/P7-25

Extension of the ECRH Operational Space with O2 and X3 Heating Schemes to Control W Accumulation in ASDEX Upgrade

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ASDEX Upgrade is operated since several years with W-coated plasma facing components. H-mode operation with good confinement has been demonstrated. Nevertheless purely NBI-heated H-modes with reduced gas puff, moderate heating power or/and increased triangularity tend to accumulate W, followed by a radiative collapse. Under these conditions, central electron heating with ECRH changes the W transport in the plasma center, reducing the central W concentration and, in many cases, stabilizing the plasma. In order to extend the applicability of central ECRH to a wider range of magnetic field and plasma current, schemes with reduced single pass absorption have been implemented: X3 heating allows to reduce the magnetic field by 30%, such that the first H-modes with an ITER-like value of $q_{95} = 3$ could be run. O_2 heating increases the cut-off density by a factor of 2 allowing to address higher currents and triangularities. In case of central X3 heating, the X2 resonance lies close to the pedestal top at the high-field side of the plasma, serving as a beam dump. For O_2, holographic reflectors have been developed which guarantee a second pass through the plasma center. The beam position on these reflectors is controlled by fast thermo-couples. Stray-radiation protection has been implemented using sniffer-probes.

EXW/P7-26

Production and Diagnosis of Energetic Particles in FAST

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The Fusion Advanced Study Torus (FAST) has been proposed as a possible European satellite facility with the aims of supporting the ITER program and of investigating physical and technological issues relevant to DEMO. One of main objectives of FAST is the study, in deuterium plasmas, of fast ion physics in conditions relevant to a burning plasma. Fast 3 He (or H) particles, with dimensionless parameters close to those of the fusion-born alphas in ITER, can be produced in FAST via 30 MW power ICRH minority heating. This work assesses to what extent these fast ion populations can be diagnosed in FAST with a set of dedicated diagnostics and what observables can be provided by each diagnostic techniques. Neutron Emission Spectroscopy (NES), Gamma-Ray Spectroscopy (GRS)

and Collective Thomson Scattering (CTS) diagnostics have been proposed for FAST and will be reviewed here with a description of the state-of-the-art hardware and a preliminary analysis of the required lines of sight. Taking as input numerical simulations of the spatial and energetic particle distribution function of the ICRH-accelerated ions for the standard FAST H-mode scenario, the corresponding spectra observed by the set of diagnostics will be simulated and their sensitivity to the fast ion tails discussed. NES and GRS measurements will provide a line averaged measurement of the tail temperature and energy density of the fast 3 He population, besides T_i with time resolutions limited by the available statistics and in the range $10 - 200$ ms. The proposed CTS diagnostics will measure the fast ion parallel and perpendicular temperature and density with a spatial resolution of 10 cm and a time resolution of 10 ms. The work will provide a scientific basis for the predictions of FAST capability in the production and diagnosis of energetic ions, and will outline the role that FAST could play in investigating fast ion physics in burning plasma experiments.

EXW/P7-27

Recent JET Experiments on Alfven Eigenmodes with Intermediate Toroidal Mode Numbers: Measurements and Modelling

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This paper reports the results of recent experiments performed on JET on Alfven Eigenmodes (AEs) with toroidal mode number (n) in the range $n = 3 - 15$. The stability properties and the use of these medium-n AEs for diagnostic purposes for the background plasma is investigated experimentally using a new set of compact in-vessel antennas, which provides in real-time a direct measurement of the modes-frequency, damping rate, mode number and amplitude.

First, we report on the development of a new algorithm for mode detection and discrimination using the Sparse Signal Representation theory. The speed and accuracy of this algorithm has made it possible to deploy it in our plant control software, allowing real-time tracking of individual modes during the evolution of the plasma background on a 1ms time scale.

Second, we report the first quantitative analysis of the measurements of the damping rate for stable $n = 3 - 7$ Toroidal AEs as function of the edge plasma elongation. The

damping rate for these modes increases for increasing elongation, as previously found in JET for $n = 0 - 2$ AEs, but contrary to measurements obtained on Alcator C-mod. Initial theoretical analysis of these data has been performed with the LEMan code. The results are in excellent agreement for all the magnetic configurations where there is only a minor up/down asymmetry in the poloidal cross-section of the plasma. These experimental results further confirm the possibility of using the edge shape parameters as a real-time actuator for control of the stability of alpha-particles driven AEs in burning plasma experiments, such as ITER. Further modelling and code-to-code benchmarks are now planned using the LEMan, CASTOR and TAEFL codes.

Finally, the diagnostic potential of medium-n AEs has being confirmed during the recent gas change-over experiments in JET, where the system has provided independent measurements of the effective plasma isotope ratio in addition to the routinely employed spectroscopic and gas-balance ones. These data shows a slight difference in the measurement of this quantity when using $n < 5$ and $n > 7$ modes, hence suggesting a radial dependence in the effective plasma isotope ratio which cannot be easily picked-up by the other diagnostic systems.

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EXW/P7-28

Reduction of Lower Hybrid Current Drive Efficiency at High Density in Alcator C-Mod

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Recent experiments on the Alcator C-Mod tokamak have shown that the population of fast electrons generated by lower hybrid current drive (LHCD), as measured by fast electron bremsstrahlung, is substantially reduced at line averaged electron densities above 10^{20} m^{-3} for 5.4 T$< B_\phi <$ 7 T. Based on results from previous LHCD experiments on other tokamaks, LHCD is expected to drive significant current with C-Mod parameters ($\omega/\omega_{lh} > 3$, $n_\parallel > n_{\parallel crit}$). Discharges with higher plasma current and magnetic field exhibit stronger fast electron bremsstrahlung emission at the same density as compared to lower current and magnetic field. H-mode discharges exhibit substantially enhanced fast electron signatures as compared to L-mode discharges at the same line averaged densities. Simulations using the GENRAY/CQL3D ray tracing/Fokker-Planck package predict a decline in fast electron bremsstrahlung similar to $1/n_e$, in line with simple theoretical estimates. Experimental results deviate from these predictions above $\sim 10^{20}$ m^{-3}, with a discrepancy of two orders of magnitude at 1.5×10^{20} m^{-3}. Wave accessibility and parametric decay into ion-cyclotron quasi-modes do not explain the lack of fast electrons at high density. Electric currents flowing between the inner and outer divertors in the scrape off layer (SOL) increase substantially as signs of fast electrons in the core plasma decrease, suggesting that absorption of the lower hybrid waves moves outside the separatrix at high density. The SOL currents during LHCD flow in the same direction as the main plasma current regardless of the launched $n_{\|}$ direction, which suggests that the current is not due to a Landau interaction between the LH waves and electrons in the SOL. By adding a SOL including collisional absorption to the numerical model, the agreement between the model predictions and experimental results improves dramatically above 10^{20} m^{-3}. Increasing the temperature of the SOL has been identified as a possible mitigation strategy to reduce absorption outside the separatrix. These results show that strong absorption of lower hybrid waves in the SOL of a high density diverted tokamak is possible and must be considered in computational modeling of future LHCD experiments.

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EXW/P7-29

A New 4 MW LHCD System for EAST

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To achieve a steady-state operation of the experimental advanced superconducting tokamak (EAST) with high parameters and to control its plasma current density profile more actively and effectively, a 4 MW lower hybrid current drive (LHCD) system will be upgraded from the present 2 MW system. It is expected to couple more than 3 MW lower hybrid power into EAST plasmas and to sustain 0.8 MA non-inductive current with a central line averaged electron density at 2×10^{19} m^{-3} and 0.6 MA at 4×10^{19} m^{-3}.

Key technologies developed for 4 MW CW LHCD system were presented in this paper, which including the power source, multi-junction antenna, 45 kV PSM (pulse step modulator) power supply, different protection (reflection and arc) and control subsystems. The experimental results are also given in this presentation for long pulse and CW operation.

EXW/P7-30

Physics and Engineering Aspects of the ICRF Heating System on EAST

Zhang, X.J.[1], Zhao, Y.P.[1], Wan, B.N.[1], Li, J.G.[1], Mao, Y.Z.[1], Yuan, S.[1], Xue, D.Y.[1], Qin, C.M.[1], Wang, C.H.[1], Lin, Y.[2], Ding, B.J.[1], and EAST Group[1]

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Experimental Advanced Superconducting Tokamak (EAST) has been built to perform advanced tokamak research in high performance regime and to explore methods for achieving a steady-state operation for a tokamak fusion reactor. Radio frequency (RF) power in the ion cyclotron range of frequencies (ICRF) will be one of the primary auxiliary heating techniques for EAST. The ICRF system provides 6 MW power in primary phase and will be capable of 10 MW later. Two 1.5 MW ICRH systems in the range of 25-70 MHz have been put into operation. ICRH system includes High-power and wide-frequency radio amplifier, the phase shifter system, the matching system and two antenna. The ICH launchers are designed to have two current straps and each strap is driven by an independent 1.5 MW RF power source. Maximum of the injected RF power reached to 800 kW using a single strap. Two ICRH systems are being employed for RF heating experiment on EAST.

Heating scheme are analyzed by using a parallel version of TORIC, a finite Larmor radius ICRF code. Three physical scenarios are considered: (a) Fast Wave (FW) minority heating; (b) FW mode conversion electron heating; (c) FW Second harmonic heating. The available diagnostic capability allows comparing experiments with the simulations and investigating different heating scenarios to address several key ICRF physical issues, such as effect of electron and ion heating, flow driven, production of energetic particles, etc.

This paper gives an overview of the main technical features of the EAST ICRH system and especially first primary results on the EAST tokamak during present campaign.

EXW/P7-31

Phased-Array Antenna System for Electron Bernstein Wave Heating and Current Drive Experiments in QUEST

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The phased-array antenna system for Electron Bernstein Wave Heating and Current Drive (EBWH/ CD) experiments has been developed in the QUEST. The incident beam had good directivity for experimental requirements in the O-X-B mode conversion scenario. The two orthogonal fields evaluated by a developed Kirchhoff code were in excellent agreement with those measured at the low power test. The elliptical polarization in two orthogonal fields can be controlled to excite pure O-mode in the oblique injection. The heat load and thermal stress in CW 200 kW operation were analyzed with a finite element code. The array phase has been fast scanned [$\sim 10^4$ degree/s] to control the incident polarization and angle to follow time evolutions of the plasma current and density. The EBWH/CD experiments in QUEST have been begun using the developed system. The non-inductive plasma current of 10 kA was ramped-up and sustained for 0.8 s at 0.133 T with only RF injection of 30 kW.

EXW/P7-32

Heat-Loads on JET Plasma Facing Components from ICRF and LH Wave Absorption in the Scrape-off-Layer

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In JET, LH and ICRF wave absorption in the SOL can lead to enhanced heat fluxes on some PFCs. In this contribution we define the location of these LH and ICRF driven hot spots on the JET PFCs, describe the dependence of the enhanced heat fluxes as a function of plasma parameters and LH or ICRH system parameters and present possible strategies to keep the temperature of these spots below critical levels in view of the operation with the new ITER-Like wall. LH absorption in front of the antenna through Electron Landau Damping of the wave with high $N_{\parallel}$ components generates hot spots precisely located on PFCs magnetically connected to the launcher. Analysis of LH hot spot surface temperature from IR measurements allows quantification of the power flux along the field lines: in worst case scenarios it is in the range of 10 MW per meter square. The main driving parameter is the LH power density along the horizontal rows of the launcher, the heat fluxes scaling roughly with the square of LH power density, with the ionisation of neutrals in the SOL by LH power playing an important role in the absorption mechanism and far SOL density enhancement. When using ICRF, hot spots are observed on the antennas structures and on limiters close to the powered antennas and are explained by acceleration of particles in RF-rectified sheath potentials. Limiters surface temperature up to 800°C at high ICRF power and high SOL density are observed. The temperature scales linearly with the SOL density and with the antenna voltage. Asymmetric wave spectra lead to a significant increase of hot spot intensity, in agreement

with antenna modelling that predicts in that case, an increase of RF sheath rectification. However, the deconvolution of IR measurements to quantify heat-fluxes is made difficult by uncertainties on the thermal models of the tiles. These models and their merit to explain surface temperature measurements are discussed. Wall protection against over heating implies the commissioning of new diagnostics (IR imaging, visible imaging, pyrometers), and the implementation of real time algorithms for hot spot recognition, and control of actuators influencing the hot spots temperature which are mainly the LH/ICRF power, the LH launcher position, and the density in front of antennas.

This work is supported by UK EPSRC and EURATOM and carried out under EFDA.

TH

Magnetic Confinement Theory and Modelling

THC — Oral

(*Magnetic Confinement Theory and Modelling: Confinement*)

THC/3-1

Towards Multiscale Gyrokinetic Simulations of ITER-like Plasmas

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Turbulence in large tokamaks spans a wide range of spatio-temporal scales, from the minor radius to the electron gyroradius, and from the transport time scale to the electron transit time. While a comprehensive first-principles treatment of all relevant effects is not yet within reach, several important multiscale phenomena with direct consequences for ITER may already be investigated in the framework of global and local gyrokinetic simulations involving multiple species, electromagnetic effects, collisions, and heat sources/sinks – using the newly extended GENE code. These include, in particular, the transport scaling with the dimensionless parameters ρ^* and β, the turbulent transport of energetic particles, and important aspects of the physics of transport barriers.

THC/3-2

Radial Electric Field Evaluation and Effects in the Core and Pedestal

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Parra and Catto [1] have point out the impracticality using quasineutrality or ambipolarity to determine the global axisymmetric electrostatic radial electric field in the core of a tokamak because an unrealistically accurate distribution function is required. We estimate the accuracy required for a direct solution of the Fokker-Planck equation to highlight why the correct radial electric field must be determined from conservation of toroidal angular momentum for both δf and full f gyrokinetic codes [2]. This procedure requires a

hybrid gyrokinetic-fluid description [2,3] that evolves the global density, temperature, and radial electric field profiles. The use of conservation of toroidal angular momentum along with radial pressure balance ensures that both the radial electric field and the toroidal flow are self-consistently determined. Full f and δf gyrokinetic equations when properly formulated automatically satisfy radial pressure balance; however, satisfying conservation of toroidal angular momentum is a more subtle issue. We also consider the strong radial electric field observed in a subsonic H mode density pedestal of poloidal ion gyroradius width, resulting in electrostatically confined ions. When the poloidal magnetic field is much smaller than the total magnetic field the poloidal $\mathbf{E} \times \mathbf{B}$ drift can then compete with the poloidal component of parallel streaming subsonically [4,5]. This competition modifies the trapped-passing boundary and shifts it onto the exponential tail of the ion distribution function. As a result, collisionless zonal flow control of turbulence is enhanced [4,5] and the banana regime ion heat diffusivity reduced [6]. Moreover, the banana regime poloidal ion particle flow can change sign [6] in agreement with C-Mod measurements [7], leading to an enhanced pedestal bootstrap current.

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THC/3-3

ITER Predictions using the GYRO Verified and Experimentally Validated TGLF Transport Model

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The trapped gyroLandau fluid (TGLF) transport model computes the quasi-linear particle, energy, and momentum driftwaves fluxes with shaped geometry and finite aspect ratio. The TGLF particle and energy fluxes have been successfully *verified* against a large database of collisionless nonlinear gyrokinetic simulations using the GYRO code. Using a new collision model in TGLF, we find remarkable agreement between the TGLF quasi-linear fluxes and 64 GYRO nonlinear simulations with electron-ion collisions. In validating TGLF against DIII-D and JET H-mode and hybrid discharges we find the predicted temperature profiles are in excellent agreement with the measured ion and electron temperature profiles. For DIII-D hybrids, the ion energy transport tends to be close to neoclassical levels while the electron energy transport tends to be dominated by short wavelength TEM/ETG modes. The high-k modes are less important in JET and ITER. ITER projections using TGLF show that the fusion gains slightly more pessimistic than the previous GLF23 results primarily due to finite aspect ratio effects included only in TGLF. The ITER results are sensitive to the improvements in the TGLF collision model while the results for DIII-D and JET hybrids are not. A new steady-state transport code TGYRO can evolve temperature and density profiles to match power and particle sources using local flux tube nonlinear GYRO simulations or a model like TGLF. TGYRO thus provides a critical *verification* of the TGLF predictions for ITER using GYRO.

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THC/3-4Ra

Hysteresis and Back Transitions in Internal and Edge Transport Barriers

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Control of internal and edge transport barriers is essential to achieving optimized confinement and performance. Successful control requires understanding of the barrier back transition mechanisms, as well as its counterpart mechanism for the barrier formation. Back transition dynamics are complicated by the phenomenon of hysteresis, whereby the barrier state persists when the driving power is lowered below the initial threshold power. Here, we report new results from computational and theoretical studies of hysteresis in transport barriers. Simulation studies using the global gyrofluid TRB code clearly show the existence of hysteresis in ion temperature gradient during power ramp-ups and downs. In addition to the gyrofluid simulations, an analysis of the barrier dynamics significantly extends previous work based on single-point S-curve bifurcation models. Progress on several critical issues in pedestal physics is also reported.

THC/3-4Rb

Gyrokinetic and Gyrofluid Simulation Studies of Non-Diffusive Momentum Transport and Intrinsic Rotation

Kwon, J.M.[1], Ku, S.[1,2], Yi, S.M.[1], Rhee, T.[3], Kim, S.S.[1], Jhang, H.[1], Terzolo, L.[1], Dif-Pradalier, G.[1,5], Diamond, P.H.[1,4,5], Chang, C.S.[1,2,4,], Kim, J.Y.[1], and Hahm, T.S.[1,6]

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Understanding intrinsic rotation and the structure of toroidal rotation profiles is crucial to optimizing plasma stability and confinement. Here, the critical tests of the theory of non-diffusive momentum transport and the results of simulation studies utilizing XGC1 and gKPSP gyrokinetic codes and TRB gyrofluid code are presented. The development of macroscopic co-current flow near the 10% of thermal velocity was observed during XGC1 simulations and the detailed physical mechanisms for the flow drive by turbulence were analyzed using gKPSP code. Also intrinsic rotation in reversed shear ITB plasmas was investigated utilizing TRB and the gyrokinetic codes.

THC/3-5

Predictions on Heat Transport and Plasma Rotation from Global Gyrokinetic Simulations

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A new generation of first principles non linear codes has recently emerged to model the turbulent and neoclassical transport in fusion plasma devices. They rely on the 5 dimensional gyrokinetic description of the plasma, which is the appropriate framework to study low collisional media. The breakthrough relies on 3 major upgrades with respect to the standard approach: the geometry is global, a large radial extent of the plasma being considered; the flux surface averaged profiles do evolve due to turbulent and neoclassical

transport; and prescribed heat sources force the system out of thermodynamical equilibrium, similarly to experimental conditions. We report on results obtained with 3 such codes (GYSELA, XGC1, ORB5) which all employ different numerical schemes to model the electrostatic toroidal branch of the Ion Temperature Gradient driven turbulence with adiabatic electrons.

Turbulent transport exhibits avalanche-like events, characterized by large scale intermittent outbursts. These bursts propagate on large radial scales at fractions of the diamagnetic velocity. For sufficiently small rhostar simulations, the shearing regions generated by the self-generated zonal flows appear to control the radial extent of the avalanches. The direction of propagation of the bursts can depend on the sign of the shearing rate.

Weak yet finite collisionality strongly impacts on plasma rotation, which is crucial for reaching enhanced confinement regimes. A twofold scan in the turbulence drive and in collisionality shows that the poloidal rotation can depart from neoclassical predictions due to turbulence. Although the turbulence generated part remains weak, it significantly increases the velocity shear, mainly through the turbulent corrugation of the mean profiles. This property increases at smaller collisionality. It may reveal important at relevant ITER parameters. Toroidal rotation decouples from the radial electric field as soon as the axi-symmetry is broken. Turbulence free simulations with magnetic ripple help in discriminating the various regimes predicted theoretically. When turbulence is active, the various processes for generation of toroidal momentum recently identified theoretically, are scrutinized. Finally, the local injection of momentum results in an avalanche-mediated redistribution of the toroidal velocity, at an effective Prandtl number of order unity.

THC/6-1

Isotope Effects on Zonal Flows and Turbulence in Helical Configurations with Equilibrium-Scale Radial Electric Fields

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Isotope effects on zonal flow generation and turbulence in magnetically-confined plasma with helical configurations are investigated by gyrokinetic theory and simulations. Poloidally global gyrokinetic simulations of the collisionless zonal flow damping manifest enhancement of the residual amplitude by the equilibrium-scale (uniform and constant) radial electric field (E_r), and show agreement with the zonal flow response kernel analytically derived from the theory for helical plasmas with multiple-helicity confinement field components. The higher zonal flow response is found with heavier ion mass for the same ion temperature (T_i) and E_r, as effects of the radial electric field are introduced in terms of the poloidal Mach number (M_p). The residual zonal flow enhanced by E_r contributes to stronger zonal flow generation by the ion temperature gradient (ITG) turbulence, and can affect the ion heat transport. Then, the isotope effects are included in the turbulent heat transport in helical plasmas with uniform E_r even without equilibrium-scale poloidal shear flows.

THC/6-2Rb

Dynamics of Low Frequency Zonal Flow driven by Geodesic Acoustic Modes

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New possible mechanism of controlling the transport is presented. Radially inward and outward propagating geodesic acoustic modes (GAMs), which is less effective on transport than stationary zonal flows (ZFs), interact to generate ZFs, which has a major impact on transport. The nonlinear dynamics of ZF driven by GAMs in tokamak plasmas is investigated theoretically, based on a fluid model. The nonlinear term (Reynolds stress) is evaluated from the action conservation equation. Truncating the nonlinearity up to the third order, the coupling equations are derived. ZF driven by GAMs is shown to be excited at high safety factor region where the turbulence driven ZF cannot be excited. The experimental parameter dependences of amplitude of ZF driven by GAMs are investigated. The parameter region where ZF is excited effectively is also pointed out. This study contributes to searching the experimental condition of new improved confinement.

THC/8-1

Role of Flow Shear Layer and Edge Plasma Turbulence in Density Limit Physics

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The fusion power scales as square of the plasma density and an achievement of a possibly highest density is a topic of great interest for economically profitable fusion reactors [1]. It is well known, however, that in tokamaks in the high confinement H-mode, there is a dramatic deterioration of plasma particle and energy confinement and back transition to the low confinement L-mode when one approaches the Greenwald limit n_{Gr} [2]. In this paper, the physical mechanism and theoretical model for the density limit in H-mode plasmas, seen normally as a back H-L transition, are presented. It is demonstrated that sheared flows, which develop during L-H transition and suppress anomalous transport at the plasma edge in the H-mode state, start to decay by generating tertiary modes like Kelvin-Helmholtz instabilities when the plasma density exceeds a critical level n_{Gr}. The dependence of n_{Gr} on the plasma current i.e., $n_{Gr} \sim I_P$ implies that for not to high currents n_{Gr} is lower than the Greenwald density limit. This offers an explanation for the experimental observations on the H-mode density limit often happening noticeably lower than n_{Gr} [2] which is actually an ultimate limit.

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THC/P2 — Poster Session

THC/P2-01

A New Steady-State Scenario for ITER based on Cyclic Operation

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A new concept of steady-state scenario for tokamak reactors is proposed [1]. It is based on cyclic operations, alternating phases of positive and negative loop voltage with no magnetic flux consumption on average. Localised noninductive current drive by Electron Cyclotron waves is used to trigger and sustain a weak Internal Transport Barrier (ITB), with a low negative magnetic shear, whereas Neutral Beam Current Drive is used to periodically recharge the tokamak transformer. The fact of operating in cycles relaxes the hard constraint of simultaneous fusion performance maximisation and full non-inductive operation. The Cronos [2] code is used to apply this concept for the ITER steady-state regime showing that the fact of combining different states opens up new possibilities to attain plasma performances that would be difficult to get in a single stationary scenario. A linear MHD analysis of the instabilities that could appear in this type of scenario is carried out with MISHKA [3] and CASTOR [4] codes, showing that MHD stability would be strongly improved with respect to a steady regime with a strong ITB as the one analyzed in [5]. Therefore, it is possible to decouple constraints that are not easily satisfied simultaneously and to alleviate MHD stability problems. Plasma regimes that are per se not suited for steady state turn out to be useful. This fact shows that, the well known problem of having different time scales in plasma physics, which has been regarded as an inherent drawback from the point of view of the plasma scenario control, can be, on the contrary, a positive feature if properly exploited.

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THC/P2-02

Physics Modeling for Steady-State Experiments at Wendelstein 7-X

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The optimized stellarator Wendelstein 7-X (W7-X) is being built in Greifswald, Germany, as a superconducting device to show the inherent steady-state capability of stellarators at reactor-relevant parameters. For this purpose W7-X is equipped with a 10 MW cw-ECRH system at 140 GHz and a High-Heat-Flux Divertor capable of withstanding 10 MW/m^2 to control particle and energy exhaust in steady-state operation. In the simplest approach, the expected plasma parameters may be derived from scaling laws like the ISS04, derived for stellarators, or for comparison from the ITER-IPB98$_{(y,2)}$-scaling assuming an equivalent circular tokamak with W7-X parameters ($R = 5.5$ m, $A = 11$, $\iota = 1$, $B = 2.5$ T). For densities of 10^{20}/m^3 and 10 MW heating power both scalings predict energy confinement times (τ_E) around 150 ms, leading to $\langle\beta\rangle$-values about 1.5%. Here, we present more detailed physics predictions based on 1D-transport modeling using a neoclassical and a simple anomalous transport model to study the accessibility of high-performance plasmas under steady-state conditions. The simulations are done for the most promising configurations of W7-X, the standard (SC) and the high-mirror configuration (HM). The SC shows an increase in tauE by a factor of up to 2.5, which makes $\langle\beta\rangle$-values of about 4% accessible for the given heating power. However, the transport simulations also show that in the SC, which has the better confinement, bootstrap currents up to 100 kA can develop while in the HM, where tauE is 30% lower than in the SC, the bootstrap current minimization from the optimization of W7-X is most effective and the net toroidal current is small. The time-evolution of the scenarios shows that on a time scale of the internal skin time (on the order of seconds) discharges in the SC without much net current may be possible. But as W7-X has no ohmic transformer, current and iota control is necessary on the global L/R-time-scale to ensure proper island divertor operation. Consequently, core and SOL-transport are combined to study the simplest scenario of a freely evolving current with a heat load pattern based on the full configuration derived from equilibrium calculations. The importance of the anomalous transport model on the results is investigated by studying ITG-turbulence in scenarios with significant T_i-gradients.

THC/P2-03

TSC Simulation and Prediction of Ohmic Discharge in EAST

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EAST is the full superconducting tokamak with D-shaped cross-section, which forms different plasma shape with elongation of $1.5 - 2.0$, triangularity of $0.3 - 0.5$, major radius of 1.8 m and minor radius of 0.4 m. The Tokamak Simulation Code (TSC), a two-dimensional time dependent free boundary simulation code, has been a very useful tool

both in analyzing experimental data and designing future tokamaks for many years. It has been used to model the time dependence of ohimc discharges in the EAST experiment for several years. The simulation results follow the experimental PF current, plasma current and position to a very good accuracy. Neoclassical resistivity gives good match with the measured surface voltage. On the basis of one good simulation, several predictions with different shape evolutions, such as elongated limiter and diverted configuration, have been done by TSC. These predictions provide reference wave (PF currents, plasma density, plasma current, and plasma position) for PCS (Plasma Control System) to direct plasma discharges. These discharges follow predictive trajectories very well. It proves the ability of TSC as predictive tool for EAST plasma discharge, and gives us a good confidence to predict 1MA ohmic discharge on EAST.

THC/P2-04

Integrated Modeling for Prediction of Optimized ITER Performance

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Detailed scenario modeling of ITER discharges is carried out using the PTRANSP predictive integrated modeling code to prepare for the commissioning of ITER and fusion burn stages of ITER operation. The PTRANSP simulation results use time evolved equilibrium plasma shape provided by the free boundary TSC code. The simulations involve three types of discharge scenarios: (1) ELMy H-mode scenarios with $I_p = 15$ MA and $n_{e,20}(0) = 1.0$ during the flat-top burn; (2) hybrid scenarios with $I_p = 12.5$ MA and $n_{e,20}(0) = 0.83$; and (3) steady state scenarios with $I_p = 8.5$ MA and $n_{e,20}(0) = 0.55-0.78$. Fusion burn simulations are carried out using deuterium-tritium fuel together with a prescribed impurity concentration (2% Be and 0.12% Ar) as well as the accumulation of helium ash from the fusion reactions. Scans for these scenarios provide insight regarding the sensitivity of the plasma behavior to uncertainties in parameters such as the H-mode pedestal height, the transport of helium ash, the impurity concentration and the effect of sawtooth frequency (in discharges with sawteeth). The minimum heating power for an L to H-mode transition is computed using the new ITPA scaling [1]. Sensitivity studies are carried out in which the H-mode pedestal height is varied around the baseline value computed using the EPED1 model [2]. Scans also include conditions that are under the control of experimentalists, such as the level of auxiliary heating power and the conditions that can be used to adjust current drive. For demanding steady-state scenario simulations, a large-bore plasma is used during the ramp-up stage, bootstrap current is maximized, and auxiliary heating is arranged to drive as much current as possible in order to maintain $q > 2$ across the plasma profile. The maximizing of fusion power is investigated by simulating reverse magnetic shear configurations and computing flow shear effects that can yield internal transport barriers. This investigation is carried out using the Multi-mode MMM08 transport model [3,4], which yields the toroidal angular frequency profile.

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THC/P2-05

Physics Based Modelling of H-mode and Advanced Tokamak Scenarios for FAST: Analysis of the Role of Rotation in Predicting Core Transport in Future Machines

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The Fusion Advanced Study Torus (FAST) has been proposed as a possible European ITER satellite. This paper presents advanced core transport modeling work in order to predict FAST operational scenarios by combining existing predictive models and most recent transport experimental results from various devices, in particular JET. A special attention has been paid to the role of rotation, which appears an essential ingredient to achieve core ion improved confinement. Rotation is modeled by assuming a turbulence driven inward momentum pinch that allows for peaked profiles even in the presence of peripheral torque sources such as due to edge intrinsic rotation or NNBI at high plasma density. The impact of rotation on plasma performance has been evaluated either according to existing transport models or according to recent JET results on ion stiffness mitigation in presence of rotation and low magnetic shear. 7.5 T / 6.5 MA standard H-modes with 30 MW ICRH, or 20 MW ICRH and 10 MW NNBI, or 15 MW ICRH and 15 MW ECRH have been simulated. 3.5 T / 2 MA fully inductive AT scenarios have also been simulated, with 30 MW ICRH and intrinsic rotation or 20 MW ICRH and 10 MW NNBI and beam driven rotation. All scenarios have been simulated using different transport models, either theory based or semi-empirical making use of recent JET experimental findings, and results critically assessed. The work has provided a sound scientific basis for the predictions of FAST performances and also for specific numerical studies of the role that FAST could play in investigating burning plasma physics as an ITER satellite.

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THC/P3 — Poster Session

THC/P3-01

On the Role of Magnetic Geometry and Flows on the L-H Transition Power Threshold

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The L-H transition power threshold, P_{LH}, is about a factor of two higher when the ion grad-B drift is away from the active X-point. As a possible explanation, the C-Mod group examined the role of SOL flows and found a correlation between them and the core rotation in L-mode. Although they argued convincingly that the SOL flows are driven by transport processes with a strong ballooning character, we offered an alternative explanation in terms of a residual vertical electric field, E_V, generated when the Pfirsch-Schluter currents cannot quite short-circuit the field due to charge-dependent drifts in a torus because of collisional effects. Symmetries of this field and the resulting flows were shown to be consistent with observations from both C-Mod, and DIII-D, which can reverse the toroidal field independently. But this point of view where the SOL flows provide a boundary condition for the core rotation in L-mode leads to incorrect expectations for P_{LH} when the effects of externally driven toroidal rotation are included. By varying the balance between co- and counter-injected neutral beams DIII-D has shown that P_{LH} is lowest with counter-injected torque and increases as one moves through balanced to co-NBI, regardless of the magnetic topology, and contrary to the expectations of this "flow point of view." The resolution of this dilemma requires switching to a view where the radial electric field is the relevant parameter. Using free-boundary equilibrium calculations, we show that the flux surface average of the radial component of E_V has a net negative (positive) value when the grad-B drift points towards (away from) the X-point. This observation alone can explain the lower P_{LH} for these cases, since the L-H transition leads to a large negative E_r at the edge, which becomes easier to attain if the average $E_r < 0$ initially. Additionally, the toroidal flow generated by co (counter)-NBI makes a positive (negative) contribution to the background E_r. Combining these observations, we expect co-injected torque when the ion grad-B drift points in the "wrong" direction to result in the highest P_{LH}. The next highest would be when the drift is in the direction of the X-point, again with co-injection. These are consistent with DIII-D results. Details of these calculations and some as yet unresolved issues will be discussed.

THC/P3-02

Coupled Core-Edge Simulations of Pedestal Formation using the FACETS Framework

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We present coupled core-edge simulations of pedestal buildup in the DIII-D tokamak using the FACETS framework. The FACETS project provides computational flexibility by allowing users to choose the best model for a given physics target. Our goals are to develop accurate transport solvers using neoclassical and turbulent fluxes with varying degree of fidelity and computational complexity, including embedded gyrokinetic simulations. Accurate sources using both ICRH wave absorption and neutral beam injection are included using parallel source components. Detailed modeling of the plasma edge using a fluid-based physics component, UEDGE, is performed and coupled to the core solver. The core region is simulated using a newly developed parallel, nested-iteration based non-linear solver while the UEDGE uses nonlinear solves from the PETSc/SNES solver package. With this new capability, simulations of DIII-D shots #118897 and #98889 are presented and compared to experiments. For this modeling, the core fluxes are computed from GLF23, NCLASS and Horton-ETG models. Sources are taken either from an interpretive ONETWO simulation or from a self-consistent NUBEAM simulation. The sensitivity of pedestal buildup to chosen transport models it studied. The effect of wall out-gassing on the core fueling and density buildup in the pedestal is discussed.

THC/P3-03

Effects of Pellet ELM Pacing on Mitigation of Type-I ELM Energy Loss in KSTAR and ITER

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Control of type-I ELMy H-mode by pellet injection has been numerically investigated for KSTAR and ITER using a 1.5D core transport code. Predictive modeling on ELMy H-mode in KSTAR successfully shows the consistent plasma dynamics compared with

experiments and provides reasonable scaling between H-mode pedestal and ELM characteristics. In order to describe the ELM triggering by pellet injection, a pellet-induced density perturbation model is proposed to carry out an ELM pacing transport simulation. The simulation results of pellet ELM pacing in KSTAR show that the pellet-induced ELMs release the reduced energy bursts compared with spontaneous ELMs, which supports the necessity of high frequency pellet injection for the mitigation of ELMs in tokamaks. The proposed methodology of pellet induced ELM triggering is applied to ITER reference ELMy H-mode scenario so that the control of type-I ELMy H-mode by pellet injection is demonstrated to check an availability of the pellet injection for the mitigation of ELM energy loss in ITER. Consequently, the optimal pellet parameters are discussed to achieve an efficient ELM control in ITER.

THC/P3-04

H-mode Transition Analysis of NSTX based on the E_r Formation Mechanism by the Gyrocenter Shift

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The radial current generated by the ion-neutral momentum exchange has been analyzed to be responsible for the radial electric field (E_r), the turbulence transport, and the low confinement mode (L-mode) to high confinement mode (H-mode) transitions on the edge of tokamak plasmas. In this analysis of gyrocenter shift the plasma pressure gradient and the neutral density gradient are the major driving mechanism of the radial current and the electric field which is formed as the source of the return current to make an equilibrium condition. When there is turbulence the small scale $\mathbf{E} \times \mathbf{B}$ eddies induce the cross-field transport. Finally the origin of turbulence is interpreted that it comes from the friction between the plasma and the neutrals so that the Reynolds number determines the state between laminar flow (H-mode) and turbulent flow (L-mode). The confinement time from the EFIT equilibrium of the national spherical torus experiment (NSTX) is compared with the density fluctuation level measured by the far infrared tangential interferometry/polarimetry (FIReTIP) to verify the turbulence induced diffusion coefficient from the theory of gyrocenter shift. In this paper theoretical explanation and the comparison with experimental data including the Reynolds number on the NSTX H-mode transitions will be presented.

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THC/P3-05

Kinetic-based Modeling of H-mode Pedestal with Theory-based Anomalous Transport Models and MHD Stability Criterion

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This study addresses the development of a scaling of the H-mode pedestal in tokamak plasmas with type I ELMs. H-mode pedestal profiles for two representative tokamak devices, DIII-D and Alcator C-Mod, are considered. The simulations in this study are self-consistent with the inclusion of kinetic neoclassical physics, anomalous transport models with the $\mathbf{E} \times \mathbf{B}$ flow shearing effects, as well as an MHD ELM triggering criterion. For the basic kinetic neoclassical behavior, the XGC0 kinetic guiding-center code [1] is used with a realistic diverted geometry. For the anomalous transport, a radial random-walk is superposed in the Lagrangian neoclassical particle motion, using the theory-based MMM95 and GLF23 models. Growth of the pedestal by neutral penetration and ionization is limited by ELM stability criterion computed by the ELITE MHD stability code [2]. The study of DIII-D discharges includes a scan with respect to plasma shaping, by varying the elongation by a factor of 1.4 and varying the triangularity by a factor of 10. The DIII-D study also includes a scan over plasma current, in which the total plasma current is varied from 0.5 to 1.5 MA with approximately fixed toroidal magnetic field, plasma shape, and normalized toroidal beta. A scaling relation for the pedestal width and height is presented as a function of the scanned plasma parameters. Differences in the electron and ion temperature pedestal scalings are investigated. The simulation results illustrate a scaling that is qualitatively similar to some experimental observations. It has been found that the pedestal width for the ion temperature profile is much wider than the pedestal width for the electron temperature and plasma density profiles. The pedestal height is found to be significantly larger in the discharges with larger elongation. Similar studies of the C-Mod pedestal are also to be reported in detail. C-Mod operates with a considerably larger value of B/R, and typically much higher absolute densities. Difference in the pedestal scaling under these conditions will be quantified and compared with empirical trends.

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THC/P3-06

Modeling of the Edge Plasma of MAST in the Presence of Resonant Magnetic Perturbations

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The impact of resonant magnetic perturbations (RMP) on the edge plasma of MAST has been studied using the B2SOLPS5.2 transport code. The effect of RMP is taken into account by adding an electron current along a stochastic magnetic field to the current continuity equation solved in the code. From analytical models it is known that the radial ion current which is generated to compensate the electron current changes the radial electric field in the separatrix vicinity and accelerates the plasma toroidally in the co-current direction. The additional particle flux which is proportional to the ion current might lead to the density drop near the separatrix - the pump out effect.

Simulations were performed for three L-mode shots with different densities with and without RMP. It was found that the electric field at the core side of the separatrix becomes less negative and plasma is accelerated in the co-current directions in all the cases with RMP. The variations in the electric field and in the toroidal rotation are consistent with those measured in these shots. A H-mode shot where more frequent type III ELMs are triggered by the application of the coils has been simulated as well. For this discharge it was demonstrated that the additional particle flux in the barrier region might explain the pump out effect and the rise of the pedestal electron temperature. For the L-mode shots the additional particle flux is not sufficient to cause a significant pump out effect. It was demonstrated by the simulations that the transport coefficients in the L-mode shots with RMP are significantly bigger than in the shots without RMP. The effect is more pronounced for the low density cases. The rise of the transport coefficients correlates with the increase of the amplitude of ion saturation current fluctuations observed in the L-mode. It was shown that the level of the magnetic field perturbations required matching the pump-out effect in the H-mode and variation of the radial electric field and toroidal rotation in the L-mode is significantly smaller than the vacuum magnetic field, i.e. significant screening of the perturbed magnetic field is predicted. The mechanisms of screening are analyzed.

THC/P4 — Poster Session

THC/P4-01

Shear Flow Suppression of Turbulent Transport and Self-Consistent Profile Evolution within a Multi-Scale Gyrokinetic Framework

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It is well known that the performance of fusion devices is limited by heat fluxes arising from small-scale turbulence driven by large-scale profile gradients. The presence of this turbulence is not inevitable, as demonstrated by the formation of transport barriers in H-mode and advanced scenario discharges on a wide range of experimental devices. The details of how these transport barriers are triggered and saturated vary from case to case, but a common feature is the favourable role played by $\mathbf{E}\times\mathbf{B}$ flow shear in providing steeper temperature gradients. Consequently, it is of great interest to determine how localized regions of strong $\mathbf{E}\times\mathbf{B}$ flow shear can develop and the extent to which this flow shear can suppress turbulent transport. We present first-principles theoretical and numerical results on the effect of flow shear on turbulence and on the subsequent evolution of the density, temperature, and flow profiles over the confinement time scale. Using the local gyrokinetic code GS2, we find that the ion thermal diffusivity is a non-monotonic function of the $\mathbf{E}\times\mathbf{B}$ shearing rate: it initially decreases with increasing shear, but then increases beyond a critical value which depends strongly on the magnetic geometry. In order to couple the microscopic fluctuation physics to the macroscopic profile evolution, we have developed a multi-scale gyrokinetic formalism and implemented it in the new gyrokinetic transport code TRINITY. We present comparisons of TRINITY density, flow, and temperature profile predictions with experimental data from a range of devices, including JET, MAST, and ASDEX Upgrade.

THC/P4-02

Global Nonlinear Gyrokinetic Simulations of Electromagnetic Turbulence in Tokamaks and Stellarators

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The theoretical understanding of mesoscale and microscale turbulence is required for developing a predictive capability of heat, particle and momentum transport in tokamaks and stellarators. Many global Particle-in-cell (PIC) codes are routinely used for solving the gyrokinetic equations in the limit of the electrostatic approximation. However, the electrostatic approximation is expected to break down in the core of high beta plasmas or in any region where pressure gradients are large. In this case, magnetic fluctuations modify the evolution of the electrostatic instabilities and eventually introduce new modes. In this paper we focus on global linear results in tokamak and stellarator geometry and we present the first global nonlinear electromagnetic simulations of finite beta effects in realistic tokamak geometry, using PIC codes. Our approach is based on an adjustable control variate method, which dramatically reduces the computational resources needed to correctly describe the evolution of the non-adiabatic part of the electron distribution function, making global electromagnetic PIC simulations of realistic plasmas finally achievable. Global nonlinear tokamak simulations are performed with a new electromagnetic version of the code ORB5. Results and benchmarks with local codes are presented and discussed, together with an analysis of the heat and particle transport in the vicinity of a static magnetic island. The same algorithms have been applied to the full 3D global gyrokinetic code EUTERPE, which uses realistic stellarator equilibria. The physical model of EUTERPE includes three kinetic species, nonlinear terms and electromagnetic perturbations. A pitch angle collision operator can also be included to allow e.g. neoclassical transport calculations or the simulation of collisional effects on TEM. Linear results and benchmarks for destabilized Alfven modes (GAE and TAE) show good agreement with existing codes. In addition to this, the linear GYGLES code has been successfully used to perform global simulations of shear Alfven modes in tokamak and pinch geometry. The fast-particle destabilization of the TAE mode and modification of the TAE into the energetic particle mode (EPM) instability (when the fast-particle drive is large enough) has been observed in the simulations.

THC/P4-03

Equilibrium and Transport for Quasi Helical Reversed Field Pinches

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In the Reversed Field Pinch (RFP) RFX-mod experiment, increasing plasma current ($I_p > 1.2$ MA) allows entering progressively better self-organized quasi helical regimes: in such regimes (named Single Helical Axis, SHAX) magnetic chaos reduction is diagnosed together with the formation of clear electron transport barriers. We provide a review of the recent progresses in theoretical/numerical studies on the related physics.

MHD analytical calculation of ohmic single helicity states by perturbation theory has been developed. A necessary criterion for field reversal at the edge is derived, proved to be satisfied in a large database of RFX-mod. Together with the study of edge radial helical magnetic field in visco-resistive MHD modelling, the indication is obtained that the validity of the necessary criterion might be more general than suggested by the perturbative approach used for its derivation: nonlinear corrections look weak, and the condition $T_e(edge) \ll T_e(center)$ sounds as a leading factor. Impact of finite beta effects will be addressed with the PIXIE3D nonlinear MHD initial value code.

The effect of chaos healing by separatrix expulsion was predicted by previous theoretical-numerical modeling. The issue remains concerning the nature of additional physical mechanisms responsible for transport in such regimes. Given previous indications of a possible role of Ion Temperature Gradient (ITG) turbulence in producing anomalous momentum transport, ITG microturbulence has been considered first. In 2008 Guo showed analytically that ITG modes are more stable in RFPs than in tokamaks because of a stronger Landau damping due to the shorter connection length. In the last two years different numerical tools, previously developed for tokamaks, have been adapted to the RFP: the nonlinear gyrokinetic GS2 code and the fluid TRB code. An integral eigenvalue approach, retaining finite Larmor radius effects has also been used. All approaches agree that ITG modes can hardly become linearly unstable, when considering ion temperature gradients similar to the electron ones. Marginal stability conditions are reached locally in the most extreme (very steep gradients) SHAX states, and might be envisaged in future higher-current experiments. Other kinds of microinstabilities are presently being studied.

THC/P4-04

Self-Consistent Simulation of Kinetic Pedestal Transport under RMP Penetration

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We report a significant new theoretical understanding of the ELM-stable edge pedestal behavior under resonant magnetic perturbations (RMPs) [1,2]. Coupled kinetic and extended MHD simulation in a realistic magnetic separatrix geometry, with electrons and ions orbiting under plasma screeing of the externally applied RMPs, self-consistent radial electric field E_r, Coulomb collisions, and neutral kinetic transport, is presented for the first time which shows that RMP amplitude is significantly shielded around the magnetic separatrix/pedestal region by plasma response, the magnetic field stochasticity survies deep into the plasma, and that n_e is significantly reduced without the collapse of T_e in a manner qualitatively consistent with experiments. The study reveals why the standard Rechester-Rosenbluth model [3] does not applicable to the tokamak edge plasmas. For the kinetic edge transport simulation, we use the full-function kinetic ion-electron-neutral guiding-center PIC code XGC0 [4]. For the RMP penetration into the plasma across the scrape-off, we use the M3D extended MHD code. It is found that a significant localscreeing of the RMP amplitude occurs around the magnetic separatrix and pedestal, and that a significant level of stochasticity survives the plasma shielding in DIII-D deep into the main plasma. Analysis of the kinetic transport in XGC0 reveals that the kinetic trapped particle effect is essential in understanding the stochastic plasma transport in a toroidal magnetic confinement device [5]. Detailed experimental validations will not be limited to the conventional tokamaks, such as DII-D and JET, but will be extended to the low-aspect devices, MAST and NSTX, which have much stronger trapped particle kinetic effects than DIII-D or JET does. Predictions for ITER plasma will also be attempted.

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THC/P4-05

Non-Local Transport Modeling of Heat Transport in LHD

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A non-local model is proposed to describe heat transport experiments conducted at the Large Helical Device (LHD). Previous successful applications of the model include the description of perturbative experiments in JET. Here we focus on recent cold pulse experiments in LHD where edge cooling was produced by pellet injection. As in previous experiments of this type, the cold pulses in LHD exhibit a time delay of the order of ~ 4 ms which is much faster than the time expected from diffusive transport. However, the key unique signature of these experiments is that the core cooling happens in the absence of significant cooling of the intermediate regions of the plasmas. For example, while a temperature drop of the order of ~ 100 eV is observed near the core, $r = 0.15$, the temperature in the intermediate zone, $r \sim 0.55$, never drops below ~ 30 eV. This tunneling-type propagation of the edge perturbation is accompanied by the existence of a long-range flux that extends throughout the whole plasma domain. Describing these phenomena using a diffusive model is problematic because the measured perturbed fluxes and temperature gradients show regions of up-hill transport (implying a negative effective diffusivity) and regions where the flux changes without a significant local change in the temperature gradient (implying a diverging effective diffusivity). In addition, the measurements indicate the lack of a single valued flux-gradient functional relation needed to justify a diffusive model with a physically meaningful spatio-temporal dependent diffusivity. We show that all this phenomenology can be described using a non-local transport model in which the total flux, q, has a component that depends on the temperature gradient throughout the whole plasma domain. Interestingly, the degree of non-locality in the model is similar to the one used in the modeling of the perturbative experiments at JET.

THC/P4-06

Nonlocal Dynamics of Turbulence, Transport and Zonal Flows in Tokamak Plasmas

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Transport and turbulence in tokamaks have traditionally been treated as local processes. We here summarise several investigations in which this "standard" paradigm is surely incomplete/incorrect, in that turbulence and transport dynamics are intrinsically nonlocal. We also report on the first observation of a novel self-organised flow structure which we call the "$\mathbf{E} \times \mathbf{B}$ staircase".

Gyrokinetic simulations using the full-f XGC1 code investigate edge-core coupling in turbulence dynamics and reveal that rapid inward propagation of turbulence intensity at a ballistic speed is accompanied by outward propagation of heat which triggers the formation of a zonal $\mathbf{E} \times \mathbf{B}$ flow shear layer in the core where the heat transport drops to near-neoclassical levels. The central importance of the pedestal region is stressed, which rather than being a mere "boundary condition" for the core, exerts on it a strong nonlocal influence, an example of how the tail (pedestal) may wag the dog (core).

Further focusing on heat conduction in the core plasma, a universal behaviour is found based on a large database from the full-f, flux-driven gyrokinetic GYSELA code: at scales smaller than a dynamically-selected meso-scale influence length, transport has the structure of a universal Levy distribution, is non-local, non-diffusive, scale-free and avalanche-mediated. At larger scales, heat transport is strongly organised into a quasi-regular pattern of $\mathbf{E} \times \mathbf{B}$ shear layers, which we call the "$\mathbf{E} \times \mathbf{B}$ staircase", a surprising tendency of the stochastic avalanches to self-organise in a jet-like pattern.

THC/P4-07

Canonical Profiles and Transport Model for Toroidal Rotation in Tokamak

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The induced toroidal plasma rotation is discussed. The equilibrium equation for rotated plasma is constructed supposing the Mach number is much less than unity. The canonical profiles of pressure and angular rotation velocity are defined as profiles, which minimize the total plasma energy with conservation of toroidal current and equilibrium condition. 2D Euler's equation is separated into three 1D equations, two of which are used to find the canonical profiles of the pressure and the angular rotation velocity w(c). It is accepted that radial flux of toroidal momentum is $q(w) = -nMR^2X(w)(w'/w - w(c)'/w(c))$, where $w' = dw/dr$ and n is the plasma density. It is supposed that the stiffness of the angular rotation velocity $X(w)$ is proportional to the stiffness of the electron temperature profile $X(e)$: $X(w) = C \times X(e)$. To find the value of C, we provide simulations of the 11 JET shots. As a result, we obtain $C = 1.0 \times n^{-1/3}$. The developed model is applied to the modelling of MAST rotation. The deviations of calculated radial profiles of w from the experimental ones for both devices are on the level of $5 - 15\%$.

THC/P4-08

Global Gyrokinetic Simulation of Electron Temperature Gradient Turbulence and Transport in NSTX Plasmas

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Global, nonlinear gyrokinetic simulations of electron temperature gradient (ETG) driven turbulence were carried out with the GTS code using actual experimental parameters of NSTX discharges. Our simulations reveal remarkable new features with regard to nonlinear spectral dynamics in 2D perpendicular wavenumber space. Specifically, there exists direct, strong energy coupling between high-k ETG modes and electron geodesic acoustic modes (e-GAMs with high frequency and poloidal mode number $m = 1$). At the same time, zonal flows are generated and continuously grow with a fine radial scale. This direct energy coupling may represent a new insight into the underlying mechanism for nonlinear ETG saturation. It also implies that the collisional damping of zonal flows and e-GAMs may have considerable impact on the formation of the steady state spectrum and saturation level. Further, the ETG fluctuation spectra are characterized by strong anisotropy with $k_r \ll k_\theta$. The $k_\perp$ spectrum of density fluctuations is in general agreement with the

experimental measurement using coherent scattering of electromagnetic waves. Sensitivity studies of simulated ETG-driven electron thermal transport with respect to the local profiles of electron temperature, safety factor and effective charge number have been carried out, given that plasma profiles and parameters are subject to significant experimental errors. Within experimental uncertainties in plasma profiles, we conclude that ETG turbulence may drive experimentally relevant transport for electron heat in NSTX. ETG turbulence spreading and its effects are also identified in our global simulations. Finally, a newly developed synthetic diagnostic, which reproduces the experimental conditions of high-k scattering, is shown to yield frequency spectra for simulated fluctuations, which are in reasonable agreement with experimental observations.

THC/P4-09

Impurity Transport driven by Electrostatic Turbulence in Tokamak Plasmas

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Impurity transport driven by electrostatic turbulence is analyzed in weakly collisional tokamak plasmas using a semi-analytical model based on a boundary-layer solution of the gyrokinetic equation. Analytical expressions for the perturbed density responses are derived and used to calculate the stability boundaries, mode frequencies, growth rates and the quasilinear particle fluxes. Parametric dependencies of the above quantities with respect to impurity charge, effective charge, impurity density scale length, and collisionality, and the effect of the impurities on the stability boundaries, have been determined and compared with quasilinear gyrokinetic simulations with GYRO resulting in very good agreement. An analytical approximate expression of the zero-flux impurity density gradient is derived and used to discuss its parametric dependencies.

THC/P4-10

Alpha Particle-Driven Toroidal Rotation in Burning Plasmas

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Alpha particle-driven toroidal rotation has been studied in burning plasmas in which the rotation effects due to external heating systems cannot be anticipated. The difference between the inward and outward drifts of alpha particles causes a steady-state ion radial current, leading to a $\mathbf{j} \times \mathbf{B}$ torque that drives the rotation. Results of simulations indicate for the first time that in burning plasmas an alpha radial current intrinsically flows outwards, but the effect of the inward movement of trapped alpha particles around

the shearless region abruptly extinguishes the current. The coupling of the OFMC and TASK/TX codes in self-consistent transport simulations enables predictions in future reactors of a quantitative toroidal rotation driven by alpha particles.

THC/P4-11

Turbulence Impurity Transport Modeling for C-Mod and ITER

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Impurity transport is clearly an important issue in toroidal magnetic confinement systems. Tokamak discharges with internal transport barriers (ITBs) have peaked main-ion profiles that may lead to higher core reactivity. However, if the impurity profile is also peaked or if impurities simply accumulate, then there will be deleterious effects on the plasma: radiation losses will increase and plasma dilution will reduce reactivity. Hollow impurity profiles are observed and hold promise for improved core performance. We address this critical area of impurity transport on ITER by using theory experiment comparison which is based on data from Alcator C-Mod and identify two important issues in understanding impurity transport. A new theory shows the impact of impurity content on nature of the drift wave turbulence and thus on the fluxes of other particles and the thermal fluxes. In addition the theoretical effect on the turbulent transport of well-structures in the radial electric field is compared with C-Mod data that has been identified as containing an internal transport barrier (ITB) form the increased gradients in the density profiles

THC/P4-12

Inward Pinch of High-Z Impurity due to Atomic Processes in a Rotating Tokamak Plasma and the Effect of Radial Electric Field

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The transport of high-Z impurity in a toroidally rotating tokamak plasma is investigated numerically and analytically. It is found that the inward pinch is driven by the atomic processes of ionization/recombination both in co- and ctr- rotating plasmas. This inward pinch is enhanced by the radial electric field. It is also found that the negative and positive radial electric fields cause the inward pinch and the outward movement (unpinch) of the high-Z impurity, respectively, under the influence of Coulomb collisions with the rotating background plasma. The pinch and the unpinch due to the radial electric field are as large as the inward pinch due to atomic processes in the case of large radial electric field.

Therefore, the inward pinch due to atomic processes and the unpinch due to the radial electric field can be suppressed by each other in the co-rotation case, while the large inward pinch occurs in the ctr-rotation case.

THC/P4-13

Impact of Toroidal Rotation on Ion Turbulent Transport in Tokamaks

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An impact of toroidal rotation on avalanche-like heat transport and momentum transport in the ion temperature gradient driven (ITG) turbulence is studied in rotation scan numerical experiments using a gyrokinetic full-f Vlasov simulation GT5D [1]. It is found that indirect influences of toroidal rotation through the mean radial electric field E_r, which is connected to parallel flows by a neoclassical force balance relation, significantly affect avalanche-like or non-local heat transport, and show asymmetric effects between co- and counter-current rotation due to a cancellation of parallel flow shear and profile shear effects. Momentum transport is analyzed for different rotation profiles, and an effective Prandtl number is estimated. The results show non-diffusive momentum transport due to E_r shear stress.

[1] Y. Idomura et al., Nucl. Fusion 49, 065029 (2009)

THC/P4-14

Plasma Size Scaling of Avalanche-like Heat Transport in Tokamaks

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The physical plasma size of future fusion devices (characterized by the normalized Larmor radius ρ^*) such as ITER will be 2 to 3 times larger than actual Tokamaks. Consequently, the understanding of the anomalous transport dependence on plasma size is one crucial issue to be addressed. Scaling laws for the confinement time assume Gyrobohm scaling, while experiments have shown that ion transport is (worse-than) Bohm, while electron transport is Gyrobohm [1]. Theoretically, several mechanisms have been proposed to explain broken Gyrobohm scaling [2], however there is no universal explanation and further investigations are needed.

In this work, the dependence of turbulent Ion-Temperature-Gradient (ITG) transport on plasma size is analyzed using the GT5D code [3]. It is studied for the first time with full-f simulations including a fixed-flux heat source and the self-consistent mean radial electric field E_r. In order to perform simulations with a larger plasma size, a straight-field-line solver [4] has been implemented in GT5D. This solver filters out the large parallel component of the wave vector in Fourier space, consistently with the gyrokinetic ordering, and does not affect the steady state of the simulations. The new findings of this work are summarized in 3 points. First, the avalanche-like heat transport, which is a dominant heat transport mechanism in the fixed-flux ITG turbulence, shows ρ^*-dependent propagation width and velocity, leading to a Bohm-like scaling. Second, the character of avalanche-like transport and its scaling is changed depending on the mean E_r shear, which is linked to intrinsic rotation profiles. This suggests a close coupling of heat and momentum transport channels. Third, intrinsic rotation profiles show weak ρ^* dependence, while density and temperature gradients are proportional to ρ^*. These complicated profile shear effects produce a transition from Bohm to worse-than-Bohm scaling.

[1] Petty et al., Phys. Rev. Lett. 74, 1763 (1995)

[2] Garbet el al., Phys. Plasmas 3, 1898 (1996)

[3] Idomura et al., Nucl. Fusion 49, 065029 (2009)

[4] McMillan et al., Comput. Phys. Commun. 181, 715 (2010)

THC/P4-15

Development of Turbulence Diagnostics on Three-Dimensional Fields Obtained by Numerical Simulations in Magnetically Confined Plasmas

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It is important to clarify the role of turbulent structures, such as zonal flows and streamer, on anomalous transport in fusion plasmas. Progress in experimental techniques of high resolution measurements enables to make quantitative estimation of turbulence transport. We have been developing a turbulence diagnostic simulator, which simulates plasma turbulence numerically. Data analyses as same in the experiments are carried out on the simulation data, and quantitative comparison, taking account of the plasma configuration, physical mechanism of diagnostics and finite spatial resolution, is made to obtain the maximum information from experimental data. In this paper, numerical diagnostics in cylindrical and toroidal plasmas are carried out. Firstly, elemental processes of mode couplings are studied in detail in the simple cylindrical configuration. A three-field reduced fluid model was extended to describe the resistive drift wave turbulence, and turbulent structures, as zonal flows and streamers have been obtained in the nonlinear saturated states. Structural formation mechanisms and their selection rule have been clarified. Applying our simulation results, we have been developing the turbulence diagnostic simulator in helical plasmas. The spatio-temporal data of turbulent fields are generated by global simulation, using a reduced MHD model describing drift-interchange instability. Global structures of fluctuations can be studied on the turbulent field. Spatio-temporal comparisons show the evidence of non-local phenomena. Numerical diagnostics are carried out on the time series of three-dimensional numerical data. We have been developing modules simulating turbulence diagnostics, such as a heavy ion beam probe and phase contrast imaging. The modules are combined with the turbulence data, and produce signals taking

account of a line of sight of each instrument with a finite spatial resolution. Quantitative comparison between turbulence and observed quantities by the numerical diagnostics will deepen our physical understanding of self-sustained structural formation mechanism.

THC/P4-16

Nonlinear Interaction Mechanisms of Multi-scale Multi-Mode MHD and Micro-turbulence in Magnetic Fusion Plasmas

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Direct numerical simulations of multi-scale multi-mode MHD and micro-turbulence at ion gyro-radius scale are performed based on gyrofluid model. We focus on the nonlinear evolution of both MHD magnetic island and micro-turbulence in a dynamically interacting system involving all zonal mode components. Here we report the progress on the understanding of nonlinear interaction mechanism with two remarkable findings: (1) A novel short wavelength ITG instability induced by a MHD magnetic island as a consequence of the breakdown of the frozen-in law. The new instability is identified to be characterized by a substantially lower stability threshold and a global structure propagating in the ion diamagnetic drift direction. (2) A magnetic island seesaw oscillation with a pivot along the singular surface due to the interaction with micro-turbulence. A minimal model is proposed to numerically illustrate the seesaw mechanism. It is identified that fluctuating electromagnetic (EM) torque due to the polarization current produced by the coupling with micro-turbulence may drive the island seesaw in the case with full reconnection. Such mechanisms offer new insights in understanding complex nonlinear interaction among multi-scale multi-mode fluctuations in fusion plasmas.

THC/P4-17

Size Scaling and Nondiffusive Features of Electron Heat Transport in Multi-Scale Turbulence

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The GTC simulations of the CTEM turbulence find that the electron heat transport exhibits a gradual transition from Bohm to gyro-Bohm scaling when device size is increased. Comprehensive analysis of radial correlation function indicates that the turbulence eddies are predominantly microscopic but with a significant component in the mesoscale.

A comprehensive analysis of kinetic and fluid time scales finds very weak nonlinear detuning of the toroidal precessional resonance of the magnetically trapped electrons that drives the linear CTEM instability. Thus the trapped electrons behave as fluid elements in the transport process, and their ballistic radial drifts across the mesoscale eddies drive a nondiffusive component in the electron heat flux. The nondiffusive electron heat flux, together with the turbulence spreading, leads to an effective electron heat conductivity dependent on the device size.

In contrast, ion heat transport in the CTEM turbulence is diffusive due to a stochastic parallel wave-particle decorrelation. The parallel wave-particle decorrelation is not operational for trapped electrons because of the bounce-averaging by the fast parallel motion. The non-diffusive feature of the electron heat transport is further conformed by the probability distribution function (PDF) of the flux-surface-averaged heat fluxes. The electron heat flux has a significant component above the lognormal distribution indicating a super-diffusive process. The ballistic electron heat flux may be responsible for the experimental observations of the residual electron transport inside the internal transport barrier (ITB) where the ion transport is neoclassical.

CTEM turbulence spreading is even more pronounced in a reversed shear tokamak. GTC simulation shows that the linear eigenmodes only appear in the normal shear region. After the saturation, the turbulence spreads across the q_{min} surface into the reversed shear region with a front propagation speed of about half of the electron diamagnetic flow. The turbulence occupies the whole volume without any identifiable gap or coherent structures in the q_{min} location or the reversed shear region. Our finding indicates that the electrostatic turbulence itself does not support linear or nonlinear mechanism for the formation of ITB in the reversed magnetic shear when q_{min} crossing an integer.

THC/P4-18

Developments in the Theory of Tokamak Flow Self-Organization

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Here we report on progress in the fundamental theory of flow self-organization in turbulent tokamak plasmas. Specifically, we discuss: a.) intrinsic toroidal rotation and the roles of residual stress, symmetry breaking and the momentum pinch. b.) turbulence driven poloidal rotation in weakly collisional plasmas. Emphasis is placed on presenting a unified theoretical framework elucidating mechanisms through which turbulence may drive toroidal and poloidal flows. Finally, we consider the coupling between poloidal and toroidal flows, and the implications of the former for intrinsic rotation studies.

THC/P4-19

Effects of Strong $\mathbf{E} \times \mathbf{B}$ Flow on Gyrokinetics

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Based on the phase space Lagrangian Lie-transform perturbation method and the field theory, a reduced kinetic model with large $\mathbf{E} \times \mathbf{B}$ flow beyond the standard ordering ($v_{\mathbf{E}\times\mathbf{B}}/v_{th} \sim O(\rho/L)$) is constructed by modifying the guiding-centre phase space transformation. Some aspects of the model are revealed and effects of the flow are discussed in course of comparison with the standard model. The push-forward representation of general particle fluid moment is presented in the subsonic flow case. In sonic flow case, corrections to the reduced quasi-neutrality condition due to the $\mathbf{E} \times \mathbf{B}$ flow are found by variational derivation of the push-forward representation of particle density.

THC/P4-20

Effects of Three-Dimensional Geometry and Collisions on Zonal Flows and Ion Temperature Gradient Modes in Helical Systems

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Effects of three-dimensional geometry of the field configuration and collisions on the zonal flows and the ion temperature gradient (ITG) modes in the Large Helical Device (LHD) are investigated by using a newly developed gyrokinetic simulation code, GKV-X. The GKV-X incorporates full geometrical information such as the Jacobian and the metric tensor of the flux surface obtained from MHD equilibrium, and it uses a novel collision operator satisfying self-adjointness and conservation laws for momentum and energy. The effects of the full geometry on the growth rate and real frequency of the ITG instability are clearly found in the large poloidal wavenumber region where the finite gyroradius effect is also important. The weak collision under the LHD experimental conditions reduces the residual zonal flow level, even if the GAM oscillation as well as the growth rate and the real frequency of the ITG modes are not much affected. Furthermore, the simulation results for the linear ITG modes are compared with a high ion temperature discharge in the LHD experiments. The poloidal wavenumbers and the critical temperature gradient of the ITG instability obtained by GKV-X show qualitative agreements with the density fluctuation measurements. The GKV-X enables ones to precisely treat the turbulent transport in high density collisional plasmas with complicated magnetic configurations, such as the edge region of the helical plasmas.

THC/P4-21

Sources of Intrinsic Rotation in the Low Flow Ordering

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We present a new, first-principles gyrokinetic model for the turbulent transport of toroidal angular momentum in the low flow ordering. This model includes new contributions that have been missed by previous models that used the high flow ordering. As a result, the intrinsic rotation becomes dependent on the specifics of the particular heating mechanism being used, and on the plasma density and temperature profiles, potentially explaining the variety of rotation profiles observed in experiments. It also recovers the momentum pinch that transports momentum from the edge into the core. The expression for the radial transport of toroidal angular momentum includes both turbulent and neoclassical contributions, and a transient term due to the change of ion pressure in time. The turbulent pieces are nonlinear terms in which the short wavelength, fluctuating pieces of the distribution function and the electromagnetic fields beat together to give long wavelength transport of toroidal angular momentum. The neoclassical contributions are integrals over the collision operator. The fluctuating pieces of the distribution function and the electromagnetic fields can be obtained by employing a higher order gyrokinetic equation that contains information about the equilibrium parallel flows. By assuming that the poloidal component of the magnetic field is small compared to the total magnetic field, we are able to formulate this gyrokinetic equation in a form that allows direct and easy implementation in existing gyrokinetic simulations.

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THC/P4-22

Nonlocal Response of Turbulent Plasma Transport in Tokamak Core on Fast Changes of Power Input

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Nonlocal response of low-frequency (LF) turbulence and the associated anomalous cross-field plasma transport on fast changes of power input in the tokamak core was simulated and analyzed. Results of four different scenarios are discussed. Basic scenario (a) illustrates plasma confinement regime in tokamak T-10 with steady ECR heating localized near a surface with minor radius $r = 12$ cm (just outside the surface $q = 1$). Two other scenarios correspond to rather quiet sawtooth-free regimes, in which the same ECR power is turned on/off (scenario (d)) or partially switched over closer to the plasma edge (scenario (b)) in particular moments of time. The simulations have shown fast (within

200 − 300 microseconds) reduction (enhancement) of anomaly heat fluxes at all radial positions after the corresponding ECR power changes in both scenarios. Scenario (c) illustrates the nonlocal response of LF turbulence and heat transport processes in the region $q > 1$ to the presence of sawtooth oscillations, which result in a fast flattening of pressure profile in the region $q < 1$ and heat release through the surface $q = 1$. Sharp bursts of the heat fluxes at all radii in the region $q > 1$ appear immediately after the flattening of the pressure profile in the region $q < 1$. The bursts indicate that turbulent convection automatically rearranges to provide the fast transfer of heat pulse through the region $q > 1$ to the edge.

THC/P4-23

Core-Edge Simulations of H-Mode Tokamak Plasmas using BALDUR and TASK Codes

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A theoretical-based model for predicting the pedestal formation of temperature and density in H-mode plasmas is used together with a core transport model in the 1.5D BALDUR and TASK integrated predictive modeling codes to simulate self-consistent simulations of H-mode plasmas. In the core plasma, an anomalous transport are computed using a semi-empirical Mixed Bohm/gyro-Bohm (Mixed B/gB), while a neoclassical transport is computed using the NCLASS model. For the pedestal, the electron and ion thermal, particle and impurity transports are suppressed individually due to the influence of flow shear. The toroidal velocity used for estimating flow shear is assumed to be a function of the local ion temperature. Because of the reduction of transport, the pedestal can be formed. As a pedestal is rising, an Edge Localized Mode (ELM) can be triggered. In this work, an ELM crash due to a pressure driven ballooning mode is considered. The combination of a core-edge model together with an ELM model is used to simulate the time evolution of plasma current, temperature, and density profiles for DIII-D and JET tokamaks. A statistical analysis will be used to quantify the predictions with experimental data.

THC/P4-24

Gyrokinetic Studies of Turbulence, Equilibrium, and Flows in the Tokamak Edge

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We report on the theory and computation of gyrokinetic turbulence in the tokamak edge. A new formulation of the gyrokinetic Lagrangian for the strong $\mathbf{E} \times \mathbf{B}$-flow regime has been found which differs from the previous version by a Lie transform. Correspondence to δf forms and to nonlinear reduced MHD was shown. This is the basis for our full f phase-space computational model FEFI. The δf model dFEFI, derived from this, in turn, is used in a comprehensive parameter study of tokamak edge turbulence in which energetic processes are quantitatively diagnosed. In this kinetic representation, edge turbulence is found to be much more sensitive to sheared flows than in previous gyrofluid studies. We explore the self consistent MHD/neoclassical equilibrium with FEFI, including the development of bootstrap flows and currents as functions of collisionality, and the collisionless control case with these effects not occurring. In both collisional and collisionless cases the conservation of particles, energy, canonical momentum, and entropy is of order 10^{-4}. We also present the status of experiment/simulation comparisons and simulated reflectometry of edge/SOL turbulence, and ongoing gyrofluid studies of ELM crash scenarios, including the influence of the bootstrap current in an edge pedestal model on both the initial instability and the resulting turbulence. These studies and findings are centrally relevant to further understanding of the H-mode and pressure profile pedestal in large tokamaks.

THC/P4-25

Intrinsic Toroidal and Poloidal Flow Generation in the Background of ITG Turbulence

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In the absence of any auxillary injection, tokamaks are believed to amplify small non-zero "seed" flows. This phenomena called "intrinsic rotation" was recently attributed to $k_\parallel$ symmetry breaking caused by $\mathbf{E} \times \mathbf{B}$ shear [1]. We report here on a different mechanism for "seedless" toroidal flow generation due to polarization drift in a fluid theoretical framework. This issue has also been recently reported as due to parallel nonlinearity driven toroidal flow in gyrokinetic framework [2,3]. The new mechanism reported here is found to be independent of the mean radial electric field shear and hence is likely to be operative in a wide parameter regime. While flow generation by the $k_\parallel$ symmetry

breaking mechanism is strongly reduced in weak pressure gradient region, the flow generation here remains unaltered, and hence is likely to be relevant in complimentary regimes. Along the same lines we obtain "seedless" poloidal flow generation due to polarization drift, which again is independent of mean radial electric field shear. We further find that turbulent momentum flux calculations in the presence of toroidal ITG introduces terms which are exclusively curvature dependent. We also demonstrate how toroidal and poloidal non-diffusive non-pinch momentum fluxes due to $\mathbf{E} \times \mathbf{B}$ shear and polarization drift get modified for toroidal ITG, as compared to slab ITG counterparts.

[1] O.D. Gürcan et al., Phys. Plasmas 14, 042306 (2007)

[2] C.J. McDevitt et al., Phys. Plasmas 16, 012301 (2009)

[3] C.J. McDevitt et al., Phys. Rev. Lett. 103, 205003 (2009)

THC/P4-26

Gyrokinetic Simulation of Temperature Gradient Instability in the RFP

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Electrostatic turbulence and transport can dominate RFP confinement when transport from global scale magnetic fluctuations is reduced by profile control and quasi-single helicity states. Ion and electron temperature gradient (ITG and ETG) turbulence and trapped electron mode (TEM) turbulence are examined in the RFP by adapting the gyrokinetic code GYRO to RFP equilibria. Solution of the Grad Shafranov equation yields toroidal generalizations of the cylindrical Bessel Function model. These are used to study instability for comparable toroidal and poloidal magnetic fields, and with the ultra low safety factor values of the RFP. RFP equilibrium parameters of importance for ITG turbulence, like magnetic shear and safety factor, are not independent and must be varied consistent with their dependence on radial position and pinch parameter. ITG modes are unstable in the RFP. Instability is enabled by the everywhere bad curvature of the poloidal magnetic field. There is no ballooning at the outside midplane. Parallel streaming is important but not enough to produce slab-like eigenmode structure in GYRO. The instability threshold in temperature gradient scale length, normalized to minor radius is comparable to the tokamak threshold scale length, normalized to major radius. This makes the critical gradient in the RFP higher by the aspect ratio. For wavelengths smaller than the sound gyroradius, nonadiabatic electron physics yields instability related to TEM and ETG. Nonlinear simulations indicate a Dimits shift as in tokamaks.

THC/P4-27

Saturation of Plasma Microturbulence by Damped Eigenmodes

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Tokamak microturbulence is shown to saturate by transferring energy to damped eigenmodes in the wavenumber range of the instability rather than by the usual energy cascade to dissipated small scales. This is established for the gyrokinetic formulation of ion temperature gradient turbulence by projecting the saturated nonlinear state onto the complete basis set of linear eigenmodes of the gyrokinetic operator. The projection is applied to initial value simulations of the gyrokinetic code GENE. For the CYCLONE base case hundreds of damped eigenmodes are excited to significant levels. Calculation of energy dissipation rates shows that the modes excited produce an energy sink in the wavenumber range of the instability that at certain scales dissipates energy faster than the instability injects energy. Proper orthogonal decomposition establishes that prompt access to dissipation in the form of weakly interacting dissipative eigenstates creates an equipartition of dissipation rate in each orthogonal mode. There are indications that damped eigenmodes impart their frequencies to the frequency spectrum at fixed wavenumber and account for the large linewidth observed in simulation and experiment. Zonal flows, while excited to large amplitude, dissipate little energy compared to other damped eigenmodes. However, energy channeled through zonal flows from the unstable mode to damped eigenmodes greatly reduces phase mixing and is calculated to be the primary energy channel for saturation. The secondary Kelvin-Helmholtz instability that breaks up the structure of the electron temperature gradient instability is found to be a dissipative structure through excitation of damped eigenmodes. Damped eigenmodes also saturate turbulence in other models of plasma microturbulence, including trapped electron mode, Rayleigh-Taylor, resistive interchange mode, and collisional drift wave turbulence.

THC/P4-28

Theoretical Transport Analysis of Density limit with Radial Electric Field in Helical Plasmas

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The study of the plasma confinement physics is the urgent task of nuclear fusion research. Phenomena of the density limit control how the plasma performance is achievable. In helical plasmas, attention has been paid to the phenomena of the density limit. The confinement property in helical toroidal plasmas is clarified. One important mechanism related with the density limit phenomena is considered to be the radiative loss of the line

emission from impurity ions. To examine the density limit due to the radiative collapse, we add the term of the radiative loss to the temporal equation for the electron temperature. The radiative loss can depend on the temperature as the loss increases if the temperature gets lower.

A set of the transport equations consists of the temporal electron and ion temperature equations and the ambipolar condition to determine the profile of the radial electric field. Since the radial electric field determined by the ambipolar condition in a non-axisymmetric system is known to affect the confinement property, theoretical analysis of the density limit needs the effect of the ambipolar radial electric field. The combined mechanism of the transport and the radiation loss is discussed. The rapid increase in the radiative loss is shown at low temperatures near the edge. The electron temperature decreases sharply near the edge and the transport barrier forms. The radial electric field takes a large negative value near the edge and the transition in the radial profile of the electric field is obtained. Because of the steep gradient in the radial electric field, the turbulent heat diffusivity reduces near the edge. If the radiative loss increases, the sum of the radiative loss and the transport loss has a minimum at the upper bound of the density and the minimum temperature to realize the stationary plasma state. The dependence of the critical density on the electron heating power and the minimum temperature in helical device is derived. The dependence of the critical density on the minimum heating power derived here is slightly stronger than the case of the Sudo scaling for the density limit.

THC/P4-29

Integrated Transport Simulation of LHD Plasmas using TASK3D

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The integrated transport simulation, in which various physical models are self-consistently connected, is essential in systematically elucidating confinement physics in toroidal plasma. We have developed an integrated transport simulation code for the helical plasma, TASK-3D, and applied to the LHD plasma. The neoclassical transport in the helical plasma is evaluated by the neoclassical transport database, DGN/LHD, where the diffusion coefficients are evaluated by the DCOM and GSRAKE codes, and the database is constructed using a neural network technique. In addition to the neoclassical transport, several anomalous transport models are included and compared against experimentally observed temperature profiles. We have implemented three turbulent transport models: the Bohm like model, Gyro-Bohm model and $1/n$ model. A constant factor for each anomalous model is determined by minimizing RMS of the differences from the measured T(e) and T(i) profiles of the LHD plasma (shot #88343, $R_{axis} = 3.60$ m, $B = 2.75$ T). The radial electric field is determined by the ambipolar condition. The time evolution of T_e and T_i are calculated using the TR module with the heat deposition profile of the experimental data. In this study we fix the density profile and the T_e and T_i profiles are

compared with the experimental data. The TASK3D results with the $1/n$ model show better agreements than the other models and the relative error between experimentally obtained T and simulation result with $1/n$ model is 4.2%. The obtained electron and ion thermal diffusivities with the $1/n$ model indicate the anomalous transport dominates in the electron thermal transport, while the neoclassical transport plays a crucial role in the ion thermal transport. The TASK/TX module, which solves the flux-surface averaged multi-fluid equation, is also applied to the LHD plasma to describe the time evolution of the radial electric field and the plasma rotation as well as the density and the temperature. The transition between the electron and ion roots, and the radial structure of the electric field have been demonstrated self-consistently.

THC/P4-30

Characteristics of Turbulence Driven Multiple-Channel Transport in Tokamaks, and Comparison with Experiments

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New features of toroidal momentum and energy transport in tokamaks found from our global gyrokinetic simulations are reported in this paper with focus on non-diffusive characteristics. i) An important nonlinear flow generation process is found due to the residual stress (a non-diffusive element of the momentum flux) produced by electrostatic turbulence of ion temperature gradient (ITG) modes and trapped electron modes (TEM). Symmetry breaking in the parallel wave number spectrum induced by turbulence self-generated low frequency zonal flow shear has been identified to be a key, universal mechanism for driving residual stress. Specifically, nonlinear residual stress generation in collisionless TEM turbulence by both the fluctuation intensity and the intensity gradient in the presence of broken $k_\parallel$ symmetry due to zonal flow shear is identified for the first time. The turbulence driven "intrinsic" torque associated with residual stress is shown to increase close to linearly with plasma pressure gradient, in qualitative agreement with experimental observations in various devices. In CTEM dominated regimes, a net toroidal rotation is driven in the co-current direction by "intrinsic" torque, consistent with the experimental trend of observed intrinsic rotation. The finding of "flow pinch" in CTEM turbulence may offer a new insight into the phenomenon of radially inward penetration of modulated flows in perturbation experiments. Simulations also suggest the existence of other mechanisms beyond $\mathbf{E}\times\mathbf{B}$ shear for $k_\parallel$ symmetry breaking. ii) Considerable non-diffusive, inward ion and electron heat fluxes associated with wave-particle energy exchange can be produced by the CTEM turbulence with typical DIII-D parameters. Moreover, the CTEM driven particle flux is shown to carry a large portion of the outward, convective flux for both electron and ion energy and toroidal momentum. iii) The ∇T_e-driven CTEM turbulence in specific parameter regimes is found to generate remarkably large fluctuation structures via inverse energy cascades, which may have a natural connection to the generation of blobs in the edge.

THC/P4-31

Gyrokinetic Simulations of Energetic Particle Turbulence and Transport

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The confinement of energetic particles is a critical issue in ITER, since ignition relies on self-heating by the energetic α-particles. In this work, the diffusion of energetic particles by microscopic ion temperature gradient turbulence is studied in large-scale simulations using a global gyrokinetic toroidal code (GTC). We find that the radial diffusivity D decreases drastically for high-energy particles due to the averaging effects of the large gyroradius and orbit width, and the fast wave-particle decorrelation. In consistent with gyrokinetic theory, GTC simulations find that energy scaling is proportional to E^{-1} for passing-particle transport due to drift-orbit averaging and wave-particle decorrelation of parallel resonance, and proportional to E^{-2} for trapped-particle transport due to gyroaveraging, banana-orbit averaging and wave-particle decorrelation of drift-bounce resonance. The gyrokinetic simulation and theory may have important implications for burning plasmas, especially for ITER. GTC simulations suggests that the transport of α-particle ($E \gg 10$) is negligible and the transport of low energy α-particle ($E \leq 10$) is relatively strong, which are good for α-particle confinement and Helium ash removal, respectively. The gyrokinetic simulation and theory also verifies the conventional concept that energetic-particle transport is reduced by gyroaveraging, drift-orbit averaging, and wave-particle decorrelation. Furthermore, energetic particle can also collectively drive shear Alfven eigenmodes in toroidal systems, such as Toroidal Alfven Eigenmode (TAE) and Energetic Particle Mode (EPM), which might induce large transport of energetic particles in burning plasmas.

Both the linear excitation (by antenna or energetic particles) of, and nonlinear transport by these toroidal shear Alfven eigenmodes have been explored through large-scale gyrokinetic simulations using GTC.

The code capability of TAE simulations has been first addressed through a series verifications by the continue spectrum in the cylindrical limit, by the gap structure in the toroidal geometry, and by the comprehensive cross-code benchmarks.

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THC/P5 — Poster Session

THC/P5-01

Distributed and Asynchronous Bees Algorithm Applied to Plasma Confinement

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There are several good stellarators configurations optimized following several criteria like diminishing the neoclassical transport or improving the stability. The search for an optimised configuration that fulfills multiple optimization criteria is a key topic to propose a reactor based upon the stellarator concept. In this work, we develop an algorithm based on metaheuristics to look for such an optimised configuration. The process is based consists of maximizing or minimizing a set of functions defined by the user, which are called objective functions. The objective functions are in our case defined on the equilibrium described by VMEC. We start by diminishing the neoclassical transport and then introduce the Mercier stability criterion, thus having two target functions presently. The ballooning stability code COBRA has also been introduced to include this criterion in the optimization process.

As introduced, the Metaheuristic algorithm explores the solution space, so a large number of executions of the fitness function is required. The best approach to perform all these computations is to use a distributed computing environment to carry out several evaluations in parallel, being ever evaluation independent from the others. A centralized master process controls the exploration of the solution space and also leads the process to optimized solutions by using improvement methods over the candidate solutions.

To our knowledge, this is the first time that metaheuristics processes have been used to solve this problem in a distributed computing environment.

THC/P5-02

3D-MAPTOR Code for Computation of Magnetic Fields in Tokamaks

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A 3D code has been developed in order to simulate the magnetic field lines in tokamaks, in two versions. In the first one, the toroidal magnetic field can be obtained from the individual fields of circular coils arranged around the torus, or alternatively, as a rippleless field. The poloidal field is provided by a given toroidal current density profile. In an upgraded version, rectangular toroidal field coils and D-shaped plasma cross sections have been included, in order to aid in the design of spherical tokamaks. Proposing initial conditions for magnetic filed lines, they are integrated along the toroidal angle coordinate, and Poincaré maps can be obtained at any desired cross section plane along the toroidal coordinate. The evolution of the field lines is also monitored from above, so the ripple due to the toroidal field coils can be appreciated. The effects of loss of axisymmetry, either originated by ripples, or by additional external coils, such as an inner coil with tilted circular loops, can therefore be studied. This is useful for the study of breaking-up of external surfaces, as in the case of ergodic divertors [1]. The code can also be used in order to reconstruct the evolution of the plasma column, using the experimental signals of tokamak discharges. In the latter case, the results have been compared with tomographic results of the ISTTOK tokamak [2].

[1] E. Chávez, J.J.E. Herrera and E. Braun, "Stochastization of Magnetic Field Surfaces in Tokamaks by an Inner Coil" in Plasma and Fusion Science, AIP Conference Proceedings Series 875 (2006) 347-349.

[2] B.B. Carvalho, et al., "Real-Time Plasma Control based on the ISTTOK Tomography Diagnostic", Reviews of Scientific Instruments 79 (2008) 10F329.

THC/P5-03

Equilibrium and Stability of High-Beta Toroidal Plasmas with Toroidal and Poloidal Flow in Reduced Magnetohydrodynamic Models

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We investigate effects of flow, finite ion temperature and pressure anisotropy on macroscopic equilibrium and stability of a high-beta toroidal plasma based on reduced magnetohydrodynamic (MHD) models. Flows may play an important role for the formation of steep structure where the scale lengths of microscopic effects cannot be neglected. A novel set of reduced equilibrium equations for high-beta tokamaks with toroidal and poloidal flow in the order of poloidal sound velocity has been formulated from two-fluid MHD

equations with ion finite Larmor radius (FLR) terms. Its solutions indicate characteristics of those effects on the MHD equilibria with poloidal-sonic flow that exhibit transition from sub- to super-poloidal-sonic flow. Stability analysis of these equilibria will also be presented. The reduced set of equilibrium equations consists of the first two orders of the Grad-Shafranov equation. This equation can be solved analytically in the limit of ideal MHD as the case for isotropic pressure. The results show that the magnetic structure is modified by the flow, the pressure isosurfaces depart from the magnetic flux surfaces due to the poloidal flow and anisotropic pressure profiles are self-consistently determined in the presence of flow. We have solved the equilibrium equations for two-fluid equilibria with flow and FLR effects numerically by using the finite element method for the case of isotropic pressure. We have obtained the following features of two-fluid equilibria: the isosurfaces of the magnetic flux, the pressure and the ion stream function do not coincide with each other, and the solutions depend on the sign of the radial electric field. Solutions for the case of anisotropic pressure will also be presented. We also perform stability analysis of the equilibria in the presence of both the toroidal and the poloidal flow obtained above. A set of reduced MHD equations that include higher order terms with time evolution have been derived. Results of linear analysis will be presented.

THC/P5-04

Three-Dimensional Equilibria with Stochastic Regions Supporting Finite Pressure Gradients

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The nature of plasma equilibria in a magnetic field with stochastic regions is examined. We show that the magnetic differential equation along chaotic field line trajectories that determines the equilibrium pressure-driven currents in the stochastic regions can be cast in a form similar to various nonlinear equations for a turbulent plasma, allowing application of the mathematical methods of statistical turbulence theory. In particular, resonance broadening theory has been applied to obtain a solution. Two difficulties must be surmounted in applying resonance-broadening theory in the context of 3D equilibria: 1) Resonance broadening theory makes use of causality, but causality does not hold in the context of equilibrium calculations; 2) The equilibrium solution in a torus must be periodic in the two angular variables, unlike the time-dependent equations to which resonance broadening theory is usually applied. In addition, we must also deal with the issue that a plasma having finite pressure gradients in stochastic regions cannot satisfy the MHD equilibrium equations. There is an extensive literature on the theory of plasma transport in the presence of stochastic magnetic field lines, and our work addresses the issue of the nature of the equilibrium solution in that context. Equilibria with stochastic regions are important for understanding fusion plasma confinement in tokamaks with ergodic limiters or resistive wall modes, or with nonaxisymmetric fields imposed for stabilizing ELMs. They are also of interest in contemporary stellarator experiments at their highest achievable values of beta, where there is evidence of the formation of a large region of stochastic field lines in the outer region of the plasma, with a finite pressure gradient in that region. The solution for the current in the stochastic region has been incorporated into the PIES 3D Equilibrium code, and has been applied to the calculation of equilibria for the W7AS stellarator. The calculated equilibrium solutions are consistent with the experimental observations, including a strong dependence of the achievable beta on the current in the divertor control coils, differences in pressure profiles between different shots in the regions calculated to be stochastic, and consistency with the Rechester-Rosenbluth estimate for the contribution of the field line stochasticity to energy transport.

THC/P8 — Poster Session

THC/P8-01

Supression of Trapped Particle Transportin Tilt Tokamaks with High Pressure Plasmas

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The results of calculations of particle drift trajectories in tilt tokamak are presented. In these calculations the model solution of Grad-Shafranov equation is used. It was shown that topology of particle drift orbits severely depends on plasma pressure and the ratio of current magnetic field to toroidal magnetic one. Turn out that these calculations the model solution of Grad-Shafranov equation is used. It was shown that topology of particle drift orbits severely depends on mutual positions of minimum magnetic field and magnetic axes. The new types of particle orbits are found. In phase space of invariants the analytical solution for the discriminant curves description is found. The qualitative estimation of banana particle quantity and the maximal deviation of drift trajectories from magnetic surface is calculated for different plasma pressures. It was shown that diffusion coefficient fall about one order of magnitude.

THC/P8-02

Dynamics of Wave-Number Spectrum of Plasma Turbulence

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Direct numerical simulations of gyro-kinetic Vlasov equation are the main tools for studying anomalous transport and thus estimating the confinement time. One of the ways the validity of these codes can be verified is by studying the wave-number spectrum of density fluctuations, which can be measured directly in a tokamak.

With this motivation, we have derived a simple model for the evolution of turbulence fluctuation spectra, which includes neighboring interactions leading to the usual two dimensional dual cascade phenomenon as well as disparate scale interactions corresponding to refraction by large scale structures.

When disparate scale interactions are dominant, which is a relevant limit for fusion plasmas, a simple spectrum for the density fluctuations of the form $|n_k|^2 \sim k^{-3}/(1+k^2)^2$ is obtained. This simple prediction is then compared to, and found to be in fair agreement with Tore Supra CO_2 laser scattering data.

In the same spirit, the wavenumber spectrum obtained from nonlinear gyrokinetic simulations of Tore Supra tokamak discharges are compared with turbulence measurements. The minor discrepancies in this comparison suggest that intermittency and non-local interactions both in space and in k-space play a role in the form of the wave number spectrum in the observed region in experiment, while in most gyrokinetic simulations such effects are either excluded or poorly treated. This shows that the physics understanding of the nonlinear interaction processes are essential for a careful quantitative comparison.

THC/P8-03

Fine Scale Zonal Flow Dynamics and Its Effect on Isotopic Dependence of Confinement

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This paper addresses the isotopic dependence of the fine scale zonal flow (ZF) dynamics which are generated from fine scale (shorter than ρ_i, but significantly larger than ρ_e) electron drift wave turbulence (DWT) which is related to either collisionless TEM or current driven drift waves (CDDW), or inverse cascade of ETG modes. We find that the fine scale zonal flows in deuterium (D) plasmas can be stronger than those of hydrogen (H) plasmas, and therefore lead to lower turbulence and transport and better confinement. We have analytically calculated the Rosenbluth-Hinton (RH) residual level of zonal flows, taking into account both ion and electron dynamics using bounce-kinetics. A salient consequence of considering two species (electrons and ions) dynamics is that the RH level shows non-monotonic behavior as a function of its radial wave vector. This happens when the neoclassical electron polarization shielding (at the electron banana width scale) begins to compete with the classical ion polarization shielding (at the ion gyroradius scale). Since the average ion gyroradius is different for different ion species for the same temperature, this transition from the ion classical polarization shielding to the electron neo-classical shielding occurs at different values of $k_r\rho_e$ for different ion species. As a consequence, RH ZFs for D plasmas can be stronger than those for H plasmas for $k_r\rho_e$ around 0.01. Based on this observation, we conclude that a significant isotopic dependence of confinement can result from the fine scale zonal flows generated from fine scale DWT if those are the dominant features of ambient micro-turbulence in particular plasmas.

THC/P8-04

Control of Turbulent Transport by GAMs

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In the tokamak edge, due to the pertinent long connection length, quasi-stationary zonal flows are strongly suppressed, and the turbulence saturation is effected by several concurrent mechanisms including geodesic acoustic modes (GAMs). GAMs could offer a tempting way to control the turbulence independent of the properties of the turbulence modes themselves. However, at present they have been experimentally observed only at rather weak displacement amplitudes compared to circular turbulence computations. It has been explored in how far flux surface ellipticity and the very high gradients encountered in the edge affect the GAM intensity.

Since experimental flux surfaces have subtle interdependences between the various geometrical parameters they are not well suitable to discriminate their effect on the GAM from the one on the turbulence. Therefore, for the numerical studies a geometry variation has been chosen which maintains the turbulence properties (growth rate, structure) at the outboard midplane as constant as possible, while varying the linear GAM properties through the global properties of the flux surfaces. That way, for example for the variation of ellipticity a strong saturating effect mediated via the GAMs can be shown quite convincingly.

In the absence of diamagnetic effects, such as for pure resistive ballooning turbulence, the diamagnetic GAM drive is switched off and the GAMs are suppressed. However for sufficiently high gradients, the GAMs return because the ratio of turbulence free energy stored in the pressure fluctuations to the turbulence kinetic energy rises, whereas the GAMs themselve maintain equal fractions of fluctuation and kinetic energy. This results in a relative rise of the GAM kinetic energy in comparison to the turbulent kinetic energy, which significantly reduces the turbulence for high gradients to intensities well below the mixing length estimate. In the limit of infinite gradients a quasistationary flow pattern results, which completely suppresses the transport.

THC/P8-05

Equilibrium Flow Shear and Magnetic Shear Effect on Zonal Flow

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A global transport simulation on ion temperature gradient driven drift wave (ITG) turbulence is performed by using electrostatic gyro-fluid code. The collapse mechanism of internal transport barrier (ITB) is investigated systematically focused on the de-correlation effect of the toroidal mode coupling in q_{min} region by flow shear which is induced by toroidal momentum source and radial electric field source, and zonal flow (ZF).

A heat source is introduced into the model and temporal evolutions of the temperature profile and the ITG turbulence is studied simultaneously in a reversed magnetic shear configuration. In the first case, the off-resonant modes below the q_{min} region are not taken into account in the simulation, where a internal transport barrier like structure is formed around q_{min} position. It is found that steady ZF is generated at the edge of rational surfaces of the toroidally coupled ITG modes.

The dependence of safety factor profile on the ZF is also studied. Reversed magnetic shear profiles with the same q_{min} value but different magnetic shear in the core region are examined. For relatively weak magnetic shear cases, the increase of the ZF around q_{min} region is observed intermittently. Especially, the ZF for the case with the weakest magnetic shear exhibits periodic oscillation. For rather strong magnetic shear case, such intermittent growth of ZF is not observed, in other words, the geodesic acoustic mode (GAM) oscillation is weak. However, the ZF is not so strong to quench the transport completely and the strong barrier formation is not observed for both cases.

In the next, the toroidal momentum source and the radial electric field source are introduced into the model independently. The effects of flow shear by these sources on the evolution of the energy propagation in spatial and wave number spaces are examined. It is found that the toroidal flow shear is not so effective for both suppression of ITG mode and toroidal coupling in the radial direction. On the other hand, it is found that $\mathbf{E} \times \mathbf{B}$ flow shear induced by the electric field source locates around q_{min} region suppresses a meso-scale structure around q_{min} and contributes to the sustainment of ITB.

The effect of off-resonance modes on turbulence suppression will be also discussed, which is now undergoing.

THD — Oral

(*Magnetic Confinement Theory and Modelling: Plasma-material interactions - divertors, limiters, SOL*)

THD/5-1Rb

Comparison between Stellarator and Tokamak Divertor Transport

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Divertor concepts presently investigated in stellarators follow the same principle as the poloidal divertor in tokamaks, aiming at similar goals and sharing the same technology. However, significant differences in divertor geometry and magnetic configuration exist between helical and axisymmetric devices, which influence the plasma, neutral and impurity transport in the SOL and consequently the functionality of a divertor. From this point of view, stellarators open a new window for exploring the optimal divertor solutions for magnetic confinement fusion devices on a broader basis.

Significant differences have been indeed observed between the poloidal divertor of tokamaks and the island-based divertors in helical devices. Using W7-AS, W7-X, LHD and AUG as examples the paper presents a comparison of the basic divertor function elements in terms of recycling, impurity retention and detachment between axisymmetric and helical devices. Physics interpretation is mainly based on EMC3-EIRENE simulations. EMC3-EIRENE is a 3D edge plasma transport code capable of self-consistently treating plasma, neutral and impurity transport in general SOLs of arbitrary geometry. The EMC3 code was initially developed for W7-AS, currently being routinely used for 3D divertor transport studies at W7-X and LHD. In addition, it has been recently applied to tokamaks like TEXTOR, DIII-D, ASDEX-Upgrade, JET and ITER.

THD/5-2Ra

Self-Consistent Integrated Modelling of Core and SOL/Divertor Transport and Simulation Study on Transient Heat Load on Divertor Targets

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Control of the power and particle exhaust is one of the most critical issues to achieve the fusion reactors, such as ITER and DEMO. To investigate the control method by the divertor, we have developed a 2D divertor code, SONIC. The SONIC suite of integrated divertor codes [1] consists of the 2D plasma fluid code (SOLDOR), the neutral Monte-Carlo code (NEUT2D) and the impurity Monte-Carlo code (IMPMC). The feature of SONIC is that Monte-Carlo approach with flexibility of modelling is applied to impurity transport. We have newly developed a self-consistent integrated modelling of core and SOL/divertor transport. Thereby it enables us to investigate operation scenarios to be compatible with high confinement core plasma and detached divertor plasmas. To integrate the 1.5D core code (TASK) and the 2D divertor code (SONIC), we introduce a new Multiple Program Multiple Data parallel computing system. For an integrated code including Monte-Carlo calculations, this system makes it possible to perform efficient simulations. The predictive simulation studies are carried out for JT-60SA with the integrated code. Dynamic change in particle and heat fluxes into the SOL region after an H-mode transition has a significant influence on divertor characteristics even in a subsequent steady state. We also investigate temporal behavior of divertor characteristics such as the transition from the divertor detachment to the attachment by the ELM heat pulse.

[1] Shimizu K., Takizuka T. et al., 2009 Nucl. Fusion 49 065028.

THD/P2 — Poster Session

THD/P2-01

Modeling of Major Disruption Mitigation by Fast Injection of Massive Li Pellets in ITER-like Tokamak Reactor

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The safest precursor of major disruption in tokamak (almost 100% of assurance) is the fast thermal quench (FTQ) of plasma center, which finishes as a fast heat shock for all tokamak plasma facing components (PFC). The moving lithium limiter created during start of FTQ by the fast injected several massive Li-pellets (for example 5 pellets with total mass $20-40$ g for ITER-like tokamak) can shield PFC from a local high power loads. Lithium noncoronal radiation and ionization losses will smooth FTQ power load in time and PFC surface. The main condition of successful shielding of PFC by this means is a moving of virtual lithium limiter from its initial position in the chamber ports to the hot plasma boundary during propagation of FTQ from plasma center to PFC (time duration of FTQ). Extrapolation showed the characteristic time development of FTQ in ITER-like tokamak can be equal to 1 ms. If the useful length between the initial position of Li-pellets in chamber ports and plasma boundary of ITER will be equal 1 m, the final Li-pellet velocity should be equal 1 km/sec. Such pellets can be accelerated by rail gun with current pulse ($30-50$ kA per pellet) cross the tokamak magnetic field. The total energy of capacitor bank used for this aim should be equal to 100 kJ. Three main questions were analyzed in paper: estimation of total amount of needed lithium, electron runaway problem and tritium removal from lithium films on chamber wall after Li-injection. The calculations of total Li and Be amount needed for ITER-like plasma cooling were based on DINA code and avalanche model of M.N.Rosenbluth and S.V.Putvinski for estimations of electron runaway. They showed the serious Li advantages. It was shown after Be injection from 3.3 up to 41 g tokamak current decay (from 14 MA to 0) finished by electron runaway generation. For runaway suppression it is needed injection of 62 g Be. In the case of Li injection for runaway suppression it is necessary only 15 g. Tritium, trapped by lithium deposit on chamber wall and LiT may be removed by chemical procedure, included the first wall exposition in vapor of D_2O and after that in CO_2 with heating of chamber up to 200°C. All hydrogen isotopes should go during these operations to gas phase and can be removed from chamber by conventional gas pumping.

THD/P3 — Poster Session

THD/P3-01

Driving Toroidally Asymmetric Current Through the Tokamak Scrape-off Layer to Control Edge-Localized Instabilities and Equilibrium Profiles

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A critical requirement for tokamak fusion reactors is the control of the divertor heat load, both the time-averaged value and the impulsive fluxes that accompany edge-localized modes (ELMs). The prediction for ITER is that ELMs will impulsively deliver extremely high heat fluxes to bounding wall components. For larger devices, even the steady-state fluxes can become unacceptably large. We propose driving toroidally non-axisymmetric current through the scrape-off layer (SOL) plasma both to broaden the SOL by driving radial convection [1] and to control the edge pressure gradient by driving resonant magnetic perturbations (RMPs) [2]. Electrostatic convection generated by direct biasing has been shown to significantly spread the SOL plasma on both MAST and NSTX. Experiments to test the effect on the divertor are currently underway on NSTX, and on MAST, this technique was shown to reduce the peak heat flux at the target plate. Experimentally, RMPs from external coils have been demonstrated to produce sufficient transport to control ELM stability on DIII-D, JET and NSTX. Calculations show that choosing the appropriate width and phasing of the biasing region at the target plate can amplify the RMP generated by the SOL current. Longer wavelength modes produce a larger effect because they are not sheared as strongly by the X-point. Generation of the necessary currents is challenging due to the possibly substantial power requirements and the possible need for internal insulators. We report on the analysis of passive current-drive mechanisms that rely on puffing and pumping of neutral gas and/or impurities in a toroidally asymmetric fashion using analytic theory and reduced 1D and 2D numerical models using the UEDGE code. The temperature at the target plate is most sensitive when the divertor is close to the transition to a highly radiative and/or attached state. Near the bifurcation point, the plasma may be able to spontaneously transition to a naturally non-axisymmetric state.

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[2] I. Joseph, R.H. Cohen and D.D. Ryutov, Phys. Plasmas 16, 052510 (2009); I. Joseph, Phys. Plasmas 16, 052511 (2009); I. Joseph, accepted by Contrib. Plasma Phys. (2009).

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THD/P3-02

Transport of Neutrals in Turbulent SOL Plasmas

Marandet, Y.[1], Mekkaoui, A.[1], Reiter, D.[2], Börner, P.[2], Genesio, P.[1], Catoire, F.[1], Rosato, J.[2], Capes, H.[1], Godbert-Mouret, L.[1], Koubiti, M.[1], and Stamm, R.[1]

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Edge plasmas of fusion devices are known to exhibit strong intermittent turbulence, which governs perpendicular transport of particles and heat. Up to now, the role of these fluctuations on the transport of neutrals in edge plasmas has not been addressed quantitatively. In fact, when coupling plasma fluid codes to neutral transport codes (e.g. B2-EIRENE [1]), the effect of turbulence on neutral transport is neglected in the sense that neutrals evolve on an average plasma background. This would remain the case even if in the future the cross-field plasma fluxes would be obtained from a direct coupling of turbulence simulations into current edge transport code suits. In this contribution, we present EIRENE calculations in a 2D slab geometry for the SOL, where the plasma background is specified by a stochastic model [2]. This model allows for calculating the effective neutral transport, averaged on the transport time scale (~ 10 ms), which is much larger than the turbulent time scale ($\sim 10\,\mu$s). The statistical properties of the density and/or temperature fields (Probability Density Function, integral scale) are specified from published experimental data [3]. We present results for the average neutral density, ionization source and Balmer line emission intensity as well as sputtering yield. We show that a fluctuating background tends to reduce the screening of neutrals as compared to the averaged background. The strongest deviations from a calculation neglecting turbulence are observed when the turbulence correlation length is large compared to the neutral mean free path. Therefore, in general, the radial profile of molecule density is more affected than its atomic counterpart. Finally, the role of fluctuations on the charge exchange flux level on the wall is investigated.

[1] D. Reiter, M. Baelmans and P. Börner, Fusion Science and Technology 47, 172 (2005).

[2] Y. Marandet, A. Mekkaoui, P. Genesio, D. Reiter, P. Börner, J. Rosato, H. Capes, L. Godbert-Mouret, M. Koubiti and R. Stamm, to appear in Contrib. to Plasma Phys.

[3] J. Boedo, J. Nucl. Mater., 390-391, 29 (2009).

THD/P3-03

Progress in Turbulence Modeling JET SOL and Edge Phenomena

Naulin, V.[1], Fundamenski, W.[2], Havlíčková, E.[3], Maszl, Chr.[4], Xu, G.[5], Nielsen, A.H.[1], Juul Rasmussen, J.[1], Schrittwieser, R.[4], and JET–EFDA Contributors[5]

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Fluid turbulence modeling of the Scrape-Off Layer (SOL) using codes that relax the usual scale separation paradigm in the SOL and treat fluctuations and time averaged background profiles on the same footing.has been an area of promising progress. Excellent progress has been achieved with the ESEL code simulating the SOL of smaller devices and the higher temperature SOL of JET. Compared to the SOL of smaller devices the SOL of JET seems to exhibit more 3D effects which are not included in the 2D nature of the original code, which presumes the absence of drift waves in the edge, and – more importantly – uses plasma streaming off with ion sound speed in the parallel direction for the SOL. We have compared several approaches to these 3D effects and numerically investigated the SOL plasma response to a source that has a complex non Gaussian PDF. The difference between the fully time-dependent approach and the stationary description of parallel transport in the SOL based on the use of time-averaged physical quantities is found to be significant. We further investigated the occurrence of coherent structures in the SOL using JET probe measurements. The origin of blobs could be fixed to the edge shear region. It is observed that blobs appear to be generated paired with a density depletion – termed a hole. While the blobs propagate outward into the SOL the holes propagate inward to the edge plasma. The lifetime of the holes is thus restricted severely due to the fast speed with which the edge plasma fills in the hole structure along the magnetic field lines. If large enough holes are arrested on resonant surfaces one can speculate that their lifetime is finite. A signature of such an event can be found in the so-called Palm Tree Mode (PTM), which is unique to JET. We were able to identify this phenomenon as a decaying current filament hole on a magnetic resonant surface, originating after an ELM event.

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THD/P3-04

Molecular Dynamics Study of Plasma Surface Interaction of Codeposited Materials

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Material mixing has attracted great interest since ITER has beryllium (Be), carbon (C), and tungsten (W) as the plasma facing components. As found in the present fusion devices, an amorphous C (a-C) layer is deposited and hydrogenated due to the bombardment with high-flux hydrogen (H) isotope ions. It drastically changes plasma surface interactions of the underlying substrate material. The interaction of low-energy H atoms with hydrogen-rich co-deposited C (a-C:H) containing Be impurity is investigated by means of molecular dynamics (MD).

An a-C layer is prepared by the bombardment of a W bcc cell by 10 eV C atoms. The H uptake in the a-C layer is performed by simultaneous bombardment with 0.025 eV C and 1 eV H atoms. Be atoms are deposited on the a-C:H layer by simultaneous bombardment with 10 eV Be and 1 eV H atoms. The bombardment of the a-C:H surface with $1-100$ eV H atoms are performed where the same initial surface is used for each bombardment. The trajectories of constituent atoms are followed for each incidence, so that the yield and the distribution of sputtered species are determined by repetitive impacts.

If there is no uptake of H in an a-C layer, CHx emission is very rare and the sputtering of C atoms shows a clear threshold for physical sputtering. By the H uptake in the layer, CHx sputtering occurs where various radicals are emitted. The H uptake induces C sputtering at energies much less than the threshold energy for physical sputtering. Both C and CHx sputtering are clearly suppressed by Be deposition on the layer. The decrease in the CHx sputtering yield is much faster than an increase of Be coverage on the surface. The strong reduction of C and CHx sputtering yields shows a good correspondence with mitigation of chemical erosion of a C target exposed to a Be-seeded plasma in PISCES-B experiments. The reason for the reduction of CHx sputtering can be explained by stronger bonding of Be-C system than for C-H and C-C bonds, leading to a formation of Be-C compound on the surface. The fraction of Be-C bonds at the surface increases steeply with increasing Be coverage, whereas the fraction of C-C and C-H bonds decreases. The results on the sputtering characteristics, which depend on the concentration of hydrogen and impurities in the codeposition, is necessary to evaluate and control the tritium inventory of the plasma facing walls in ITER.

THD/P3-05

Advances in Understanding Tokamaks Edge/Scrape-off Layer Transport

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Progress is presented on understanding plasma/neutral transport issues for tokamaks, from both moment-equation and kinetic-equation descriptions. For fluid modeling, the focus is on time-dependent and cross-field drift effects using UEDGE as the primary simulation tool. Results are presented for neutral fueling of the pedestal during the ELM cycle, plasma particle and energy transport beyond the secondary separatrix for blob transport, and cross-field drift algorithms with plasma flow and electric field effects. Applications are made to DIII-D, ITER, and a Snowflake divertor configuration. Kinetic transport results are presented from the continuum TEMPEST code [4D, $(2r, 2v)$] for combined collisional and anomalous radial transport in divertor geometry. Verification tests are shown for the new COGENT 4D code that utilizes state-of-art 4th-order conservative finite volume numerical scheme to implement a conservative formulation of the gyrokinetic equation. The tests include particle trapping, ion acoustic modes, and 2D geodesic acoustic modes (GAMs), which give expected quantitative behavior including T_e/T_i scaling for GAMs. Future development of COGENT for 5D is discussed. A consistent, implementable 2nd order gyrokinetic equation set is also presented. The set retains only necessary terms, and is good for electromagnetic simulations, and various approximations are proposed. To understand plasma/wall boundary conditions more deeply, $(2r, 3v)$ particle simulations are presented that resolve orbit shapes and closing of diamagnetic currents at walls.

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THD/P3-06

Model Microfield Method in Atomic Kinetics of Turbulent Thermonuclear Plasma

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Plasma parameters (such as temperature and density) in modern magnetic confinement devices are subject to strong fluctuations due to the effects of plasma turbulence. At the first edge and divertor tokamak plasmas the scales of turbulent fluctuations are comparable to plasma parameters themselves. Under such conditions the kinetics of atomic states

populations is changed essentially leading to uncertain interpretation of plasma medium properties such as atomic spectral lines intensities and radiative losses.

The effect of plasma turbulence on atomic kinetics depends on a ratio of typical turbulence fluctuation times to relaxation times in an atom. In the present work the well known in spectral lines broadening theory model microfield method (MMM) is used for the atomic kinetics description in the general case of arbitrary relation between the atomic relaxation and the plasma turbulence times. From the viewpoint of atomic system evolution the turbulent fluctuations (for example, temperature) are described by the amplitude of the fluctuations and their frequency. The both parameters are the input data in MMM. The method is based on an approximation of turbulent noises by Markoff process with a specific distribution of amplitudes and a frequency (jumps of plasma parameters). It is possible to obtain an analytical solution of a system of linear kinetic equations for atomic states populations influenced by such perturbations. This solution may be used for the atomic states population evolution analyses in turbulent plasmas.

The case of atomic lithium in edge plasma was considered as the simplest example of atomic kinetic problem solution. The solution obtained depends on atomic relaxation constants (ionization and recombination rates) as well as on plasma parameters fluctuation frequency. The investigation of the system evolution in the limiting cases of slow and fast fluctuations enables us to make a conclusion about the atomic population evolution both at fixed value of plasma parameters with subsequent averaging over their distributions and the evolution for initially averaged values of these parameters. For the particular model of static distribution of fluctuating plasma parameters the effects of its action on atomic kinetics may be calculated in terms of solution by MMM.

THD/P3-07

Study of Radial Particle Transport Accompanied with Plasma Blob and Self-Organized Meso-Scale Structure in Tokamak Scrape-off Layer

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The non-diffusive radial transport in the SoL is investigated using the 2D SoL interchange turbulence simulation code, which reproduces plasma blobs. From the nonlinear simulations, it is found that the meso scale convective cells emerge from interchange instabilities and propagate in the radial direction. The structures are formed by the inverse cascade of the convective turbulence autonomously. Based on this phenomenon, the radial propagation velocity of the blob structure and the effective SoL radial convective transport velocity are compared. They agree well each other. The effective transport coefficient is also introduced and the Bohm-like dependence is found.

THD/P3-08

Validation of Turbulent Plasma Transport Simulations for Collisional Linear Plasma

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Significant progress has been made with 3D simulations of plasma turbulence in the linear plasma experiment LAPD [1], with successful comparison to a range of experimental data. The level of overall agreement is unprecedented for edge plasma turbulence simulations. Not only order-of- magnitude consistency but a detailed match is observed for a range of characteristics: temporal spectra, spatial spectra, and fluctuations size probability distribution. Thus developing confidence in relevance of the model, we use these simulations to get physics insights on important aspects of plasma transport relevant to the boundary of fusion devices. It is recognized that there is universality of key features of turbulent plasma transport for scrape-off layer plasmas in rather different devices: tokamaks, stellarators, and linear machines, in particular LAPD. Consistent with that, our modeling results indicate that similar to fusion-relevant edge plasmas, the transport is predominantly carried by large events ("blobs"); this is remarkable in the absence of magnetic curvature. A detailed analysis of simulated turbulent transport in the linear plasma will be presented and physical mechanisms will be discussed.

[1] Gekelman et al., Rev. Scientific Instrum., 62, 2875-2883 (1991).

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THD/P4 — Poster Session

THD/P4-01

Neoclassical Approach to Angular Momentum Transport and Toroidal Rotation in Tokamak Plasmas

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Approaching the problem of toroidal momentum evolution within the framework of the neoclassical theory with the corrected Braginskii stress tensors, this paper looks for the counterparts of terms like fluxes and sources in the momentum conservation equations and presents their numerical solutions for the evolution of the toroidal and poloidal velocities. Present study considers a subsonic, collisional plasma in front of the magnetic separatrix having a model temperature profile with a controllable gradient and a pedestal height. Study indicates a nonlinear, two-time-scales-coupling between the poloidal and toroidal rotation velocities and shows that the poloidal rotation velocity has a faster response time. If gyrostress tensor is properly taken into account, however; the longer-time evolution of the poloidal and toroidal rotation velocities are strongly coupled. This behaviour is found to be governed by a system of three quasilinear partial differential equations where the space variable is a radial boundary layer distance from the magnetic separatrix. Possibility of a solution is determined by the chosen initial and boundary conditions, (Dirichlet or Neumann), at both limits of the radial boundary layer and the gradient and pedestal height of the model temperature curve used. Steep temperature gradients are found to lead to rapidly diverging rotation velocity profiles.

THS — Oral

(*Magnetic Confinement Theory and Modelling: Stability*)

THS/1-1

A First Principles Predictive Model of the Pedestal Height and Width: Development, Testing, and ITER Optimization with the EPED Model

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This effort develops, tests, and utilizes a predictive model for the H-mode pedestal height and width based upon two fundamental and calculable constraints: 1) onset of non-local peeling-ballooning modes at low to intermediate mode number, 2) onset of nearly local kinetic ballooning modes at high mode number. The pressure at the top of the edge transport barrier (or "edestal height") strongly impacts global confinement and fusion performance. Accurately predicting the pedestal height in ITER is an essential element of prediction and optimization of fusion performance. Investigation of intermediate wavelength MHD modes (or "peeling-ballooning" modes) has led to validated understanding of an important constraint on the pedestal height and the mechanism for ELMs. Systematic calculation of peeling-ballooning stability bounds, with the ELITE code, provides a constraint on the pedestal height, as a function of the edge barrier width. The EPED models employ local onset of the kinetic ballooning mode (KBM) as a second constraint. The KBM constraint is developed using a "ballooning critical pedestal" technique, in which an edge barrier profile is taken to be ballooning critical when the inner half of it is at or beyond the local threshold. The EPED1 model used a simple parameterization, emphasizing the dominant dependence of the pedestal width on the value of poloidal beta at the top of the pedestal. The new EPED1.5 model calculates both the peeling-ballooning and KBM constraints directly for each case. This yields a prediction which is fully first principles, in the sense that no parameters are taken from observations, and takes into account additional dependencies in the KBM relation. The EPED model has been extensively and successfully tested, including dedicated experiments on DIII-D and C-Mod with predictions made prior to the experiment, and tests on JET and JT-60U. Initial tests of EPED1.5 find a ratio of predicted to observed pedestal height of 0.97 ± 0.19 in a set of DIII-D and JET discharges. Predictions for ITER from both EPED1 and 1.5 yield a high pedestal ($\beta_{N,ped} \sim 0.6$), and are used to explore optimizations of both pedestal, and, in conjunction with core studies, global performance in ITER.

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THS/4-1

High-Beta Physics of Magnetic Islands in 3D Equilibria

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Numerical simulation and analytic theory is used to describe the effects of finite plasma pressure on magnetic island formation and magnetic surface fragility in three-dimensional geometry.

The extended MHD code NIMROD is used to simulate high-beta equilibria in net current-free straight stellarator geometry using a resistive MHD model with anisotropic heat conduction. The calculation proceeds initially from a 3D vacuum magnetic field that has rotational transform. The dominant symmetry of the initial vacuum magnetic field can be spoiled through the application of small 3D helical harmonics. A heating source is introduced in order to form finite beta equilibria. Anisotropic heat conduction is employed with typical ratios of parallel to perpendicular heat conduction $\chi_\parallel/\chi_\perp \sim 10^5 - 10^8$. Since parallel heat conduction is finite, pressure gradients are allowed to persist in regions with small magnetic islands or magnetic stochasticity. A connection between flux surface destruction and the breaching of MHD stability boundaries is investigated.

Analytic calculations of magnetic island formation in 3D equilibria employ drift kinetic theory to describe self-consistent current responses. Three dimensional components of $1/B^2$ produce net radial drifts that give rise to viscous forces on the plasma and neoclassical cross-field transport. In tokamak physics, this is conventionally referred to as neoclassical toroidal viscosity. In addition to the dissipative response, there is a reactive component to 3D fields that describes in-surface currents. These currents self-consistently affect magnetic island formation through the perturbed parallel current response generated by the quasi-neutrality condition $\nabla \cdot \mathbf{J} = 0$. The strength of this response depends upon the relative role of collisions, $\mathbf{E} \times \mathbf{B}$ and precessional drifts. The reactive response counteracts the island producing effects due to resonant Pfirsch-Schlüter currents.

This work suggests that high-beta stellarators are more resilient to retaining flux surface integrity at high temperature than predicted from conventional resistive or neoclassical-MHD theory.

THS/7-1

Non-Linear MHD Simulations of Natural and Pellet Triggered ELMs

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In recent years, many detailed measurements of Edge Localised Modes (ELMs) have become available. Direction-linear MHD simulations of ELMs have advanced to the point where comparisons of the ELM simulations with detailed experimental observations can be envisaged. Qualitatively, ELM simulations with the non-linear MHD code JOREK show a good agreement. The formation of filaments of density, observed in fast Thompson scattering measurements, is due to the ballooning mode convection cells. The fine structure, observed in the energy deposition at the target during ELMs, is found to be due to the magnetic field perturbation due to the ELM. The large parallel energy conduction along the magnetic field perturbed by the ballooning mode leads to a spiral pattern on the target plates. Also the movement of the stripes in time is well reproduced by the MHD simulations.

In order to advance towards more quantitative comparisons with experiment, ELMs are simulated in JET equilibria using the EFIT equilibrium reconstruction and high resolution Thomson scattering for the edge pedestal profile data.

One option for the control of the large ELM energy losses foreseen in ITER is the injection of pellets. Each injected pellet triggers an ELM, i.e. the ELM frequency and thereby the ELM size can be controlled by the frequency of the pellet injection. The origin of the trigger of an ELM by the pellet is not yet clear. Non-linear MHD simulation of the pellet injection into an H-mode pedestal close to marginal stability suggest that the high pressure inside the high density plasmoid generated by the pellet may trigger a ballooning-like mode. This mode is characterized by a single dominant filament formed by a coupling of many toroidal harmonics.

THS/9-1

Sawtooth Control Relying on Toroidally Propagating ICRF Waves

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For the first time it is verified that fast ions generated from ion cyclotron resonance heating (ICRH) have a direct impact on sawtooth destabilisation. Dedicated JET experiments have been devised in order to test the predictions of a recent theory [1] indicating that sawteeth can be destabilised by ICRH in reactor relevant conditions. Energetic passing ions influence the MHD internal kink mode instability when they are distributed asymmetrically in parallel velocity, which is a natural feature of minority ion populations in resonance with toroidally co or counter propagating ICRF waves. Reported here are theory and dedicated JET experiments which have been devised in order to neutralise an alternative sawtooth control mechanism, involving changes in the equilibrium current due to ICRF, and permit comparison with recent theory across physical parameters. Negligible change to the net equilibrium current was assured by choosing ITER relevant 3He minority ICRF. Depending on the antenna phasing, the sawtooth period can remain as short as those of Ohmic pulses, despite 7 MW of auxiliary heating. Sawteeth were also controlled with these techniques in H-mode. A change of antenna phasing can produce sawteeth that are so long that NTMs are triggered and saturated even in L-mode. The sawtooth stability properties are explained by the effect of the wide drift orbits of fast ions intersecting the $q = 1$ radius. Employing the SELFO generated distribution function in the drift kinetic code HAGIS reveals the corresponding fast ion contribution to the potential energy. The latter characterisation of stability has been shown to agree well with an analytical solution involving the fast ion parallel and perpendicular temperatures, and the gradient of the fast ion current at $q = 1$. In summary, that fast ions can so dramatically, and directly, affect sawteeth is encouraging for ITER, especially where control solely via the magnetic shear is expected to be more challenging.

[1] J.P. Graves, et al, Phys. Rev. Lett. 102, 065005 (2009)

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THS/9-2

Runaway Electron Modeling for Rapid Shutdown Scenarios in DIII-D, Alcator C-Mod, and ITER

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Runaway electron confinement during disruptions in three tokamaks – DIII-D, Alcator C-Mod, and ITER – is modeled with a newly developed extension to the 3D MHD code NIMROD. Experiments in C-Mod focus on the de-confining effects of disruption-induced MHD during massive gas injection (MGI). The NIMROD simulations, in agreement with experiment, find that during MGI the seed runaway population is entirely lost. In DIII-D, $n = 3$ magnetic perturbations are applied to de-confine runaways experimentally and in the simulations. NIMROD finds that these $n = 3$ fields can interact with and qualitatively alter the disruption-induced MHD, but has no direct effect on confinement of runaway in the core. Simulations of ITER are conducted with parameters that are realistic for the current quench phase. These simulations allow direct comparison with the smaller tokamaks of runaway electron confinement results under similar disruption scenarios, allowing the effects of larger machine size, and higher temperature and plasma current to be considered.

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THS/P2 — Poster Session

THS/P2-02

Parametric Study of Equilibrium and Stability Analysis of HT-6M Tokamak in the Presence of Flow

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It is experimentally observed that tokamak plasmas exhibits macroscopic flow in the toroidal and poloidal directions. Both the flow can considerably change the equilibrium parameters of tokamak. Equilibrium of HT-6M is revisited as a case study and its equilibrium parameters are simulated in the presence of toroidal and poloidal flows with the FLOW Code developed by L. Guazzotto at that University of Rochester (USA). FLOW code was originally designed for spherical tokamaks, which is modified to implement on the circular tokamak.

Effect of toroidal and poloidal flow on poloidal flux, current density, particle density, temperature, beta profile, toroidal and poloidal magnetic fields for isotropic and anisotropic cases is studied. A comparison is also made for no flow and with different flows. It is found that the plasma is squeezed against the outboard side of the tokamak producing an outward shift (Shafranov shift) because of centrifugal force. It is also observed that q-profile is modified in the presence of flow.

The presence of flow also pose problem with the stability of toroidal system. For our analysis of stability we studied effect of different poloidal and toroidal flows on internal kink modes. Presence of internal transport barrier is not only considered good for steady-state operation but it also effects the equilibrium of the tokamak especially q-profile (safety factor). Therefore effect of transport barrier in the presence of toroidal and poidal flow on equilibrium and stability are studied.

THS/P2-03

Onset and Saturation of a Non-Resonant Internal Mode in NSTX and Implications for AT Modes in ITER

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Motivated by experimental observations of apparently triggerless tearing modes, we have performed linear and nonlinear MHD analysis showing that a non-resonant mode with toroidal mode number $n = 1$ can develop in the National Spherical Torus eXperiment (NSTX) at moderate normalized β_N when the shear is low and the central safety factor q_0 is close to but greater than one. This mode, which is related to previously identified "infernal" modes, will saturate and persist, and can develop poloidal mode number $m = 2$ magnetic islands in agreement with experiments. We have also extended this analysis by performing a free-boundary transport simulation of an entire discharge and showing that, with reasonable assumptions, we can predict the time of mode onset. These instabilities are also found in linear simulations of ITER hybrid and AT regimes, which are being extended to include the onset of energetic particle modes.

THS/P2-04

Real-Time Control of Neoclassical Tearing Mode in Time-Dependent Simulations on KSTAR

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Neoclassical tearing mode (NTM), a type of resistive instability, is one of major concerns in present and future tokamaks aiming at achieving plasmas with high fusion performance. Based on theoretical analyses, the time evolution of NTM can be interpreted and predicted by the modified Rutherford equation (MRE). The method using localised current drive to replace the missing bootstrap current with electron cyclotron current drive (ECCD) is expected to be one of the most promising ways for active stabilisation of the NTMs due to its narrow deposition width and equipment of the real-time steerable mirror. However, it is difficult to align ECCD to NTM accurately because of modification of the ECCD deposition and the island location in time according to evolution of the plasma transport and equilibrium. In this work, real-time feedback control of NTM is simulated using a transport code, ASTRA, coupled with the MRE at KSTAR. The plasma equilibrium, transport and island evolution is calculated self-consistently. Based on the results, performance of the NTM control is expected to be highly improved by optimising alignment

between the island and the ECCD in both time and space. This will strongly contribute to ITER as well as forthcoming KSTAR experiments.

THS/P2-05

KSTAR Equilibrium Operating Space and Projected Stabilization at High Normalized Beta

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The Korea Superconducting Tokamak Advanced Research, KSTAR, is designed to produce steady-state high beta plasmas to establish the scientific and technological basis of an attractive fusion reactor, and to provide research supporting advanced scenario operation of ITER. Projecting active and passive stabilization of global MHD instabilities for operation above the ideal no-wall beta limit is therefore an important study. Along with an expanded evaluation of the equilibrium operating space, experimental equilibria of the most recent plasma discharges were reconstructed using the EFIT code. Near-circular ohmically-heated plasmas created in 2009 have reached stored energy 66 kJ, normalized $\beta = 0.88$, and plasma internal inductance 1.0. Plasma current has reached 0.34 MA. Kinetic modification of the ideal MHD $n = 1$ stability criterion was computed by the MISK code on KSTAR theoretical equilibria with plasma current of 2 MA, internal inductance of 0.7, $\beta_N = 4.0$, and DIII-D H-mode pressure profile shape. The steep edge pressure gradient and high temperature cause a large ion diamagnetic frequency and a large negative edge $\mathbf{E} \times \mathbf{B}$ frequency, resulting in the need for significant plasma toroidal rotation to allow thermal particle kinetic resonances to stabilize the RWM. The KSTAR design includes a mode control system comprised of passive stabilizing plates and a uniquely designed set of in-vessel control coils (IVCC). The impact of various materials and electrical connections of the passive plates on RWM growth rates was analyzed. Copper plates reduced the RWM passive growth rate by a factor of 15 compared to stainless steel at $\beta_N = 4.4$. Computations of active RWM control using the VALEN code show that the $n = 1$ mode can be stabilized by the midplane IVCC at β_N near the ideal wall limit. The ELM mitigation potential of the IVCC, examined by evaluating the island overlap created by resonant magnetic perturbations (RMP), is analyzed using the TRIP3D code. Using all IVCCs with dominant $n = 2$ field and upper/lower coils in an odd parity configuration, a Chirikov parameter near unity at normalized poloidal flux 0.85 was generated in theoretical high beta equilibria. Chirikov profile optimization is examined versus β_N and safety factor profile. Optimizing the RMP using higher-n spectra is also addressed for possible IVCC upgrades.

THS/P2-06

Wall Forces Produced during ITER Disruptions

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A very critical issue for the ITER device construction is to evaluate the forces produced on the surrounding conducting structures during plasma disruptions. The asymmetric sideways force produced by kink modes in ITER is calculated in simulations and theory. New numerical diagnostics are derived and implemented, in which the jump in the magnetic field across a thin resistive boundary gives the wall current, which multiplied by the magnetic field, yields the wall force. The present simulations are based on a ITER reference equilibrium, which is unstable to a vertical displacement event (VDE), and is rescaled to become unstable to an external kink. The coupled VDE and kink mode are evolved nonlinearly. The wall force, like the current and pressure, is quenched by interaction with the wall. The wall force is somewhat dependent on wall penetration time. The wall force is largest when the mode growth rate is of order of the reciprocal of the wall penetration time. A theory is developed of the wall force produced by kink modes. The theory is in qualitative agreement with the simulations. In particular, the theory and simulations give positive correlation of sideways force with the sideways component of the kink displacement, and asymmetric vertical force with vertical kink displacement. A net toroidal variation of the plasma current is produced in the presence of both a kink mode and a VDE, which is correlated with the kink displacement.

THS/P3 — Poster Session

THS/P3-01

Mechanisms of the Plasma Rotation Effect on the Type-I ELM in Tokamaks

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Mechanisms of the plasma rotation effect on the type-I ELM related MHD modes are investigated numerically. To clarify them, a new energy is defined, which relates to the potential energy without the contribution of the Doppler shift. By resolving this energy into each component that includes the rotation effect on the equilibrium and that on the displacement of the MHD mode, it is found that the MHD mode is destabilized by the difference between the eigenmode frequency and the equilibrium toroidal rotation frequency. This destabilizing mechanism is essentially induced by the rotation with steep shear, and becomes effective as the wavelength of the MHD mode becomes shorter. Based on these results, a toroidal rotation effect on the type-I ELM is investigated numerically in JT-60U H-mode plasmas. This analysis shows that the sheared toroidal rotation can have impact on the achievable pressure gradient and type-I ELM behavior in JT-60U by changing the MHD stability.

THS/P3-02

Integrated Simulation of ELM Triggered by Pellet through Energy Absorption and Transport Enhancement

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Two integrated core / scrape-off-layer (SOL) / divertor transport codes TOPICS-IB and JINTRAC with links to MHD stability codes have been coupled with models of pellet injection to clarify effects of pellet on the ELM behavior. The energy absorption and the transport enhancement by the pellet were found to trigger the ELM. The ablated cloud of pellet absorbs the background plasma energy and causes the radial redistribution of pressure due to the subsequent $\mathbf{E} \times \mathbf{B}$ drift. On the other hand, the sharp increase in local density and temperature gradients in the vicinity of ablated cloud causes the transient enhancement of heat and particle transport. Both mechanisms produce a region of an increased pressure gradient in the background plasma profile within the pedestal, which triggers the ELM. The mechanisms can explain a wide range of experimental observations.

THS/P3-03

Turbulent Dynamics of Li-seeded Tokamak Plasmas at the Edge

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Simulations of turbulent tokamak plasma dynamics at the edge have been performed. Four-field $(\phi, n, p_e, V_\parallel)$ nonlinear MHD model has been analyzed. Two-liquid reduced Braginsky set of equations has been solved for the hydrogen plasmas. The total set of equations included neutral hydrogen and lithium atoms, and lithium ions with the charge $z = 1$. Radial distribution of radiation losses, and concentrations of lithium species (Li^0, and Li^+) have been calculated for different concentrations of neutral hydrogen and Lithium atoms at the wall. Calculations show that the penetration of hydrogen and Lithium neutrals into the low temperature edge plasma layer decrease the electron temperature as a consequence of radiation losses. Ionization of neutrals increases the electron density. Plasma flows rise significantly if the wall concentrations of hydrogen atoms rise. For example, the wall neutral hydrogen density increasing from 0 to 4×10^{11} cm^{-3} the particle turbulent flow increases by factor 2.5. The influence of the lithium wall concentration isn't so strong. For example, the lithium wall concentration rising from 0 to 2.5×10^{11} the particle turbulent flow increases by factor 1.4.

THS/P3-04

Magnetic X-Points, Edge Instabilities, and the H-Mode Edge

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A new picture of the edge of D-shaped toroidal fusion plasmas emerges from recent large scale nonlinear MHD simulations of Edge Localized Modes (ELMs) and applied non-axsiymmetric fields. A freely moving magnetic boundary surface possessing one or more X-points, surrounded by vacuum, undergoes asymptotic surface splitting when perturbed, similar to a two-degree of freedom Hamiltonian dynamical system with a hyperbolic saddle point. The magnetic field lines develop two different limiting surfaces, or manifolds, when traced infinitely in opposite directions. The unstable manifold forms loops around the original surface, or helical field-aligned lobes on the 3D torus, while the stable manifold remains close to the original surface. The multiple intersections of the two manifolds and their connecting field lines create a stochastic magnetic tangle. In an ELM, given an initially unstable ballooning-type instability on the outboard side of the plasma, magnetic stochasticity first involves the X-point regions and can then destabilize the originally stable inboard side of the torus. The instability, together with the magnetic tangle, can propagate deeply into the plasma through an interchange, rather than magnetic tearing mechanism. Field lines remain locally aligned with the equilibrium field. Most are confined over many toroidal transits before being lost from near the X-points. Significant pulsed losses of density and energy can occur directly to the divertors from near the X-points. The underlying magnetic configuration is preserved and the plasma gradually relaxes back toward axisymmetry and the original boundary. At realistic and near-realistic resistivity, the resulting multi-stage ELM matches many experimental observations. Field splitting provides a natural mechanism for generating a weakly stochastic magnetic field that is consistent with a steep edge gradient region sustained by a minimum heating power, as observed in H-mode. The theoretical picture has connections to recent developments in fluid turbulence.

THS/P3-05

Nonlinear ELM Simulations based on Peeling-Ballooning Modes using the BOUT/ BOUT++ Code

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A minimum set of equations based on the Peeling-Ballooning (P-B) mode with non-ideal physics effects (diamagnetic drift, $\mathbf{E}\times\mathbf{B}$ drift, resistivity, and anomalous electron viscosity) is found to simulate pedestal collapse when using the BOUT++ simulation code, developed in part from the original fluid edge code BOUT. Linear simulations of peeling-ballooning modes find good agreement in growth rate and mode structure with ELITE calculations. The influence of the $\mathbf{E}\times\mathbf{B}$ drift, diamagnetic drift, resistivity, and anomalous electron viscosity on peeling-ballooning modes is being studied; we find that (1) the diamagnetic drift and $\mathbf{E}\times\mathbf{B}$ drift stabilize the peeling-ballooning mode in a manner consistent with theoretical expectations; (2) resistivity destabilizes the peeling-ballooning mode, leading to resistive peeling-ballooning mode; (3) anomalous electron viscosity destabilizes the peeling-ballooning mode, leading to a viscous peeling-ballooning mode. With addition of the anomalous electron viscosity under the assumption that the anomalous kinematic electron viscosity is comparable to the anomalous electron thermal diffusivity, it is found from nonlinear simulations using a realistic high Lundquist number that the pedestal collapse is limited to the edge region and the ELM size is about $5-10\%$ of the pedestal stored energy. This is consistent with many observations of large ELMs. It is also shown that for high Lundquist number there are two distinct processes in the evolution of pressure profiles: a fast collapse greatly flattening the pressure profile near the peak pressure gradient on the order of tens of Alfven times after the onset of nonlinear P-B mode and a slow buildup of pressure gradient. We can characterize the fast collapse as a magnetic reconnection triggered by P-B modes $\rightarrow$ an island formation and magnetic braiding $\rightarrow$ bursting process and a slow collapse as a turbulence transport process. The estimated island size is consistent with the size of fast pedestal pressure collapse. The ELM size and power distribution in the Scrape-Off-Layer from nonlinear simulations will be compared to DIII-D experiments and to ITER design/operation scenarios.

THS/P3-06

Plasma Response to Externally Applied Resonant Magnetic Perturbations

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The response of tokamak plasmas to applied resonant magnetic perturbations (RMPs) are studied numerically by using the nonlinear two-fluid equations. It is found that: (1) For a sufficiently small RMP amplitude such that the RMP is not fully penetrated, the plasma rotation is speeded up by a static RMP in case the original rotation was in the ion diamagnetic drift direction with a frequency smaller than the electron diamagnetic drift frequency, but it is slowed down otherwise. For sufficiently large RMP amplitudes, the plasma rotation frequency approaches the value of the electron diamagnetic drift frequency but rotates in the ion drift direction. (2) A sufficiently large RMP is found to flatten the local electron density profile around the rational surface as expected. In case of a small or moderate RMP amplitude, however, the electron density is increased if the plasma rotates in the ion drift direction with a frequency being larger than the electron drift frequency. In the opposite limit the electron density is decreased, and its local radial gradient may even become positive for a given constant perpendicular particle diffusivity. (3) There is a minimum required field amplitude for mode penetration if the mode frequency is the same as the externally applied RMP frequency. The mode penetration threshold increases as the mode frequency deviates from the RMP frequency and is asymmetric on the two sides of the minimum value. In case of a plasma rotation velocity faster than the electron diamagnetic drift (but rotating in the ion drift direction), the required field amplitude is larger. (4) For low mode frequency, the mode locking threshold of magnetic islands is found to be proportional to the plasma viscosity and mode frequency but inversely proportional to the square of the Alfvén velocity and the island width. The low-m islands are predicted to be locked by a very small amplitude error field in ITER or a fusion reactor. In addition to the above mentioned results, two fluid simulations for particle transport in stochastic magnetic fields will be presented.

THS/P4 — Poster Session

THS/P4-01

Reversal of Impurity Pinch Velocity in Tokamak Plasma with a Reversed Magnetic Shear Configuration

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Impurity transport in tokamak core plasmas is investigated with a three dimensional fluid global code. The diffusion coefficient and the pinch velocity of impurity transport in tokamaks is studied using fluid model for ion temperature gradient (ITG) and trapped electron mode (TEM) driven turbulence in tokamak plasmas. It is shown that in presence of an internal transport barrier (ITB) created by a reversed magnetic shear configuration, one can obtain the reversal of impurity pinch velocity which can change from inward direction to outward direction. This scenario is favorable for expelling impurities from the central region and decontaminating the core plasma. The mechanism of pinch reversal is attributed to a modification of the dominant underlying instability caused by a change of the gradient of the ion temperature consequently of the transport barrier formation.

THS/P4-02

Integral Torque Balance in the Problem of the Plasma Toroidal Rotation

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The study is aimed to clarifying the balance between the sinks and sources in the problem of the intrinsic plasma rotation in tokamaks recently reviewed in [1]. The integral torque on the toroidal plasma is calculated analytically using the most general MHD plasma model with account of plasma anisotropy and viscosity. The contributions due to several mechanisms are separated and compared. It is shown that some of them, though, possibly, important in establishing the rotation velocity profile in the plasma, may give small input into the integral torque. This gives a key to the judicious choice of the directions of necessary studies. The role of the boundary conditions in the problem is discussed. This is a step to relate the plasma rotation to the physical characteristics measured outside the plasma. The analysis shows that an important contribution can come from the magnetic field breaking the axial symmetry of the configuration. In stellarators, this is a helical field which is needed for producing the rotational transform. In tokamaks, this can be the error field, the toroidal field ripple or the magnetic perturbation created by the correction coils in the dedicated experiments. The estimates for the error-field induced torque show that the amplitude of this torque can be comparable to the typical values of torques introduced into the plasma by the neutral beam injection. Therefore, this torque must be considered as an important part of the integral torque balance. The obtained relations allow to quantify the effect that can be produced by the existing correction coils in tokamaks on the plasma rotation, which can be used in experiments to study the origin and physics of the intrinsic rotation in tokamaks. Several problems are proposed for theoretical studies and experimental tests.

[1] deGrassie J.S., Plasma Phys. Control. Fusion 51 (2009) 124047.

THS/P5 — Poster Session

THS/P5-01

Progress in Understanding the Multiscale Analysis of Magnetic Island Interacting with Turbulence in Tokamak

Agullo, O.[1], Muraglia, M.[1], Voslion, T.[1], Benkadda, S.[1], Guimaraes-Filho, Z.O.[1], Garbet, X.[2], Yagi, M.[3,4], Sen, A.[5], Waelbroeck, F.L.[6], Caldas, I.L.[7], Nascimento, I.C.[7], and Kuznetsov, Y.K.[7]

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We report on progress in understanding the multi-scale dynamics of magnetic islands in presence of turbulence. In tokamak, many instabilities can generate small-scale turbulence but it is not clear how they can affect the large scale magnetic island dynamics and which mechanisms involve the coexistence of both. In this work, we focus on two different kinds of small-scale turbulence, namely pressure and flow driven ones. The pressure driven turbulence develops in the vicinity of the resonance and therefore perturbs the island dynamics while the second one takes place between the two magnetic islands because of the existence of flow gradients in this region, a common situation in Advanced ITER scenarii Discharges. We'll show that turbulence can generate seed magnetic islands which are known to be precursors for neoclassical tearing modes (NTMs). NTMs are known to limit the level of plasma pressure in magnetically confined plasmas and are therefore an important issue for a fusion reactor such as ITER. In this work, we'll focus on the underlying mechanisms which link turbulence and seed islands.

THS/P5-02

Impact of the Geometry of Resonant Magnetic Perturbations on the Dynamics of Transport Barrier Relaxations at the Tokamak Edge

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The dynamics of transport barrier relaxation oscillations in presence of resonant magnetic perturbations (RMPs) is investigated with three-dimensional turbulence simulations of the tokamak edge. The results share common characteristics with edge localized mode (ELM) control experiments. In particular, the stabilizing effect of RMPs on barrier relaxations is reproduced qualitatively. Such stabilization combined with practically no degradation of the energy confinement is due to modifications of the geometrical properties of the barrier. These modifications result from different competing mechanism, i.e. convective transport associated with stationary structures linked to residual magnetic islands as well as an increase of collisional radial transport and a decrease of turbulent transport associated with field line ergodization. The competition of these mechanisms as well as the impact of different geometries of the magnetic perturbation (multiple harmonics, single harmonic, position of the resonance with respect to the barrier) on the control of the barrier relaxations and the quality of the confinement are investigated.

THS/P5-03

Tearing Modes in Electron Magnetohydrodynamics

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The dissipation mechanism of collisionless reconnection is analyzed, and the effects of anisotropy pressure gradient and guide field gradient on tearing mode are also analyzed in electron magnetohydrodynamics. It is found that either pressure-based dissipation or inertia-based dissipation dominates, has a great relation with the relative scaling orders between the electron thermal Larmor radius and electron inertia skin depth. The effects of pressure gradient also depend on the relative magnitude between parallel and perpendicular equilibrium pressure gradients. When the pressure-based dissipation is dominant, the condition that pressure drives or suppresses tearing mode instability also depends on the relative magnitude between parallel and perpendicular equilibrium pressure gradients. It is also shown that the guide field gradient has a significant influence on tearing mode. When the guide field gradient is smaller than the magnetic field shear at the magnetic null plane, the growth rate of tearing mode instability is enhanced by the guide field gradient and has no oscillatory component. When the guide field gradient is larger than

the magnetic field shear, the guide field gradient can destabilize tearing mode instability dramatically. In this case, the growth rate is proportional to the guide field gradient.

THS/P5-04

Response of a Resistive and Rotating Tokamak to External Magnetic Perturbations Below the Alfvénic Frequency

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Plasma response to magnetic perturbations from the RWM to TAE frequency range is studied by using the MARS-F code to minimize the free energy of the plasma and excitation coils. MARS-F takes into account plasma toroidal flow, resistivity and/or various kinetic effects and an arbitrary external coil geometry. The present work was motivated by the discovery that magnetic perturbations can stabilize the ELMs. Previous systematic and useful studies on ELM suppression using the SURFMN code neglected the intrinsic plasma response. It led to the conclusion that outer $\sim 10\%$ flux surfaces were found to be stochastic and posed difficulty to observations that plasma edge remains in the H-mode with good confinement. In this work, we verified SURFMN by checking it against analytic models and also MARS-F assuming vacuum plasma conditions. With inclusion of plasma, with flow profiles and various resistivity levels, MARS-F response deviates from vacuum significantly for the resonant components of perturbations. For ideal-plasma, the resonant components are completely suppressed at the mode resonant surfaces. With plasma resistivity, this suppression (or shielding) becomes imperfect. However, with experimentally measured rotation profile, and with a wide range of plasma resistivity (magnetic Reynolds numbers $S = 10^6$ to 10^8), the shielding remains substantial. In comparison to the vacuum response, the size of the magnetic island, which is proportional to $\sqrt{B_n}$, is much reduced and the field line stochasticity is limited to the outer $\sim 2\%$ of the flux surfaces. This is consistent with the observation of only very minor modification to plasma transport. Similar results were found for different plasma shapes with various plasma elongation and triangularity. At higher frequencies, which are relevant for the TAE, RSAE etc, for plasma responses with large kinetic energy δK, its frequency is found independent of the geometry of the external coils, but the amplitude could be very different. The width in frequency of the response peaks is related to the continuum damping, which is also obtained by adding plasma resistivity. We conclude that the present formulation and results can be extended to study the perturbation of plasma by external coils in future devices, such as ITER.

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THS/P5-05

A Numerical Matching Technique for Resistive MHD Stability Analysis

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We have developed a new numerical matching technique for linear stability analysis of resistive magnetohydrodynamics (MHD) modes. The conventional asymptotic matching theory has several difficulties in the numerical implementation in practice because of the singularity of the problem; i.e., (i) it is not applicable to the case where the minimum safety factor being a rational number in reversed-magnetic-shear tokamak plasmas since the rational surface becomes irregular singularity, and (ii) the numerical implementation is not so straightforward because of the sensitive behavior on the local equilibrium accuracy and grid arrangements near the rational surface. Unlike the conventional asymptotic matching theory, our numerical matching technique utilizes an inner layer with a finite width, which resolves those practical difficulties in the asymptotic matching theory. Neither the radial coordinate nor the growth rate is expanded by using smallness of the electrical resistivity in the inner layer. The inner-layer solution is directly, not asymptotically, matched onto the outer-region solution. We found that this direct matching is accomplished by requiring the continuity of perturbed magnetic field and the smooth disappearance of the parallel electric field. Our matching method is successfully applied to single and double tearing, internal kink and interchange modes.

THS/P5-06

Non linear, Two Fluid Magnetohydrodynamic Simulations of Internal Kink Mode in Tokamaks

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The non-linear, two fluid magnetohydrodynamic (MHD) code XTOR-2F [1] is used to investigate the dynamics of sawtooth cycling in tokamaks. The equation set evolved is a subset of the bi-fluid model that can be derived from the Braginskii equations. The only two-fluid effects considered are the diamagnetic drifts. In the current work, we describe the ramp phase of sawtooth oscillations, and how the diamagnetic drift terms affect the regimes where sawtoothing can occur. Non-linear, 2 fluid MHD simulations including resistivity and transport, are carried out in plasmas that approximate the conditions found in ohmically heated low performance tokamak discharges. Non-linear simulations are carried out at poloidal beta $\beta_p = 0.11$, 0.22, and 0.44, with and without diamagnetic rotation effects. The threshold for the ideal kink mode lies at about $\beta_p = 0.35$. The non-linear simulations yield different internal kink regimes. There is a β_p threshold for oscillations – below this threshold, plasmas show periodic kink oscillations; while, above the threshold, plasmas evolve toward a three dimensional equilibrium with a saturated $m/n = 1/1$

kinked core. The inclusion of ion or electron diamagnetic drifts increases the β_p at which kink cycling can occur. Moreover, a sawtooth crash cycle as observed in tokamak experiments is recovered in the simulations, including precursor and postcursor instabilities. In some cases, cancellation of the diamagnetic stabilization may occur, in which case the background resistive MHD behavior is restored, accelerating the reconnection rate of the kink instability and decreasing the crash time. It is found that the ratio of the resistive time to the sawtooth period, obtained in the simulations, is about a factor of two away from typical Ohmic experiment measured sawtooth periods.

[1] H. Lütjens et al., Plasma Phys.Control.Fusion Vol.51, 12 (2009), 124038.

THS/P5-07

Effects of Turbulence Induced Viscosity and Plasma Flow on Resistive Wall Mode Stability

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The effect of a new dissipation mechanism, turbulence induced viscosity, on the resistive wall mode (RWM) stability is studied in the cylindrical plasma. The eigenmode equation for RWM is derived, including the turbulence induced viscosity, the plasma flow and the parallel viscosity. The eigenmode equation is solved numerically. Test computations are carried out to study the dependence of the mode growth rate on the wall conductivity, for a case without viscosities and without plasma flow. In the absence of flow but with the turbulence induced viscosity and parallel viscosity, numerical results indicate that the growth rate of the RWM decreases quickly with enhancement of the turbulent viscosity. In the presence of the plasma flow, the results show that the mode is completely suppressed as the plasma rotation frequency exceeds a critical value. It is also found that, when the mode starts to be stabilized, the real frequency of mode tends to saturate and finally decays. The numerical results also exhibit that the turbulent viscosity significantly reduces the critical rotation frequency required for the RWM stabilization. The effect of the turbulent viscosity on the stability window, in terms of the wall minor radius, has also been investigated.

THS/P5-08

Multi-Scale MHD Analysis Incorporating Pressure Transport Equation for Beta-Increasing LHD Plasma

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In the Large Helical Device (LHD) experiments, high beta plasmas have been successfully obtained in the unstable configuration against linear ideal interchange modes. We investigate the stabilizing mechanism utilizing a nonlinear simulation based on the reduced MHD equations. Since the linear stability property strongly depends on the beta value, we develop a multi-scale simulation scheme including beta-increasing effects, which treats both time evolutions of perturbed and equilibrium quantities with different time scales simultaneously. In the scheme, the equation for the average pressure is separately treated as a pressure transport equation, which consists of nonlinear convection, diffusion of background pressure and continuous heating source. This equation governs the dynamics of the background pressure. The time evolution of the LHD plasma in a beta-increasing phase is examined by means of the scheme. The heating source in the transport equation is adjusted so that the beta value should increase in a constant rate. The numerical result shows a stable evolution of the LHD plasma with weak fluctuations. At low beta, the interchange modes are weakly excited and saturates immediately because the driving force is small. The weak excitations generate local flat regions around the resonant surfaces in the background pressure profile. Such structure decreases the driving forces of the modes at higher beta. Therefore, the local reduction of the background pressure gradient due to the nonlinear dynamics is considered to be the stabilizing mechanism of the LHD plasma. On the other hand, the generation of a flat region enhances the pressure gradient in the vicinity. The enhancement of the gradient has a destabilizing contribution to a mode resonant at the region. Therefore, there are interactions between multiple unstable modes with different helicities through the deformation of the pressure profile. As a result, the plasma is continuously self-organized as a whole in the increases of beta so that vortices induced by all of the modes should not overlap significantly. The result obtained here indicates the existence of a stable path to a high beta regime obtained in the experiments. Relation between the nonlinear evolution and the linear stability and convective transport due to the nonlinear dynamics are also discussed.

THS/P5-09

A Mechanism of Structure Driven Nonlinear Instability of Double Tearing Mode in Reversed Magnetic Shear Plasmas

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The nonlinear destabilization of the Double Tearing Mode (DTM) and the subsequent collapse have attracted much attention as a crucial problem which terminates high performance reversed magnetic shear plasmas, but the underlying physical mechanisms, especially the trigger mechanism, have not been solved. Here, we found a possible process which is characterized by two distinct but coupled secondary instabilities with two different time scales, i.e. A) a secondary instability which triggers the growth of potential flows with faster time scale, and B) a zonal field driven tearing instability which triggers that of the magnetic flux with slower time scale. The first one originates from the 2D magnetic islands deformation during the quasi steady-state nonlinear regime, which may result from modulational process, whereas the latter emerges from the current corrugations, i.e. the zonal field, generated by the coupling between destabilized flows and slowly evolving magnetic flux. These two mechanisms are subsequently coupled, leading to the formation of strong kinetic flows and to the full reconnection/collapse.

THS/P5-10

Modelling of Plasma Response to RMP Fields in MAST and ITER

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The linear full MHD code MARS-F, its drift-kinetic extension MARS-K, and the nonlinear reduced free-boundary MHD code BOUT++ are used to compute the plasma response to the applied magnetic fields, produced by the resonant magnetic perturbation (RMP) coils for the purpose of the type-I ELM mitigation in H-mode tokamak plasmas. The linear, full toroidal modelling yields quantitative prediction of the steady state plasma response (the magnetic field and the plasma displacement) at a fixed plasma rotation. The non-linear modelling allows self-consistent evolution of the plasma rotation, hence being very useful in understanding the field penetration dynamics.

We find that, as expected, the ideal/resistive plasma response near the plasma edge significantly modifies the resonant spectrum of the vacuum field, produced by the RMP coils. The plasma response reduces the field amplitude near rational surfaces, forming a deep "valley" in the spectrum diagram. A rapid toroidal plasma rotation can provide a

very efficient shielding of the field, bringing the resistive plasma response close to that of the ideal response. With the expected rotation speed for the ITER 15MA ELMy H-mode plasma, the amplitude of the total radial field at rational surfaces is $2-3$ orders of magnitude smaller than that predicted by the vacuum calculations in the plasma inner regions, and about 10% of the vacuum field in the plasma edge region. Decreasing the plasma rotation speed increases the plasma response everywhere, in particular in the inner regions. At the rotation speed close to the expected value for ITER, the plasma surface displacement is about 1mm per kA of the $n=-4$ current, in a wide range over the poloidal angle near the LFS. The displacement increases rapidly with decreasing the rotation speed.

The computed plasma response field is essential in estimating the magnetic islands' width, the islands' overlapping and the possibility of the eventual field line ergodisation. Work is going on to study the effects of diamagnetic flow, the two fluid response, and the drift kinetic physics on RMP response, using linear and nonlinear codes.

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THS/P5-11

Effects of Toroidal Geometry and FLR Nonlocality of Fast Ions on Tearing Modes in Reversed Field Pinch

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Strong ion heating and plasma momentum transport are observed during periodic magnetic relaxation events in self-organized high temperature plasmas such as the reversed field pinch (RFP). The cycle is characterized by a slow growth phase followed by a short "crash" phase in which magnetic reconnection rapidly alters the field, transfers magnetic energy to ion thermal energy, and modifies mean plasma flows. Two types of tearing modes drive impulsive reconnection in the Madison Symmetric Torus (MST) RFP experiments: $m=1$ modes in the plasma core and $m=0$ edge modes resonant at the reversal surface where the equilibrium toroidal field changes its sign. It is seen experimentally that when $m=0$ modes are not present there is also no change in stored magnetic energy, ion temperature, or toroidal plasma rotation. Motivated by the special importance of this mode for RFPs, we report here the first computational results for an edge-resonant, $n=1$ tearing mode in a toroidal geometry in combination with nonlinear simulations of its cylindrical analog and new analytic and numerical results for finite Larmor radius (FLR) stabilization of tearing modes by neutral beam injected (NBI) fast ions. The key findings are: (1) the structure of the toroidal $n=1$ edge resonant mode is substantially different from the cylindrical $m=0$, $n=1$ mode and agrees well with experimentally measured profiles; (2) cylindrical single fluid and two fluid MHD computations show a strong axial squeezing of the $m=0$ magnetic island caused by the nonlinear coupling

of a large number of edge resonant modes with $m = 0$ and $1 < n < 20$; (3) analytical solutions obtained in the asymptotic limit of large Larmor orbits predicts a significant stabilizing effect of fast ions on tearing modes in the MST NBI experiments; numerical modeling using NIMROD with a pressure tensor calculated from the particle-in-cell method demonstrates good agreement with the analytical results.

THS/P5-12

Sub-grid scale Effects on Short-wave Instability in Magnetized Hall MHD Plasma

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Nonlinear evolutions of short-wave unstable modes have been studied to clarify the saturation mechanisms of pressure-driven instabilities in the Large Helical Device, LHD[1,2]. Two numerical results are shown in this article.

Firstly, we see numerical results of full-3D MHD simulations of LHD. The numerical simulations of LHD are carried out by the use of the MINOS code [1] for the initial equilibrium of the inward-shifted magnetic axis position 3.6 m and the peak beat 3.7%. We show that the unstable modes can be saturated mainly through both linear and nonlinear contributions of the parallel heat conductivity, and the saturated profile of the pressure should be studied more correctly by an extended-MHD model. We also find in the simulations that the azimuthal wavelength of the high ballooning modes are comparable to the ion skin depth, and that we can expect further stabilization in two-fluid effects. Here we consider modeling the sub grid scales of the high-n modes so that we can carry out LES of an extended/two-fluid MHD equations in which scales of interests are solved sharply while emergence of very fast waves at higher modes can be eliminated.

Secondly, in order to compromise a time-consuming nature of a 3D extended/two-fluid simulation due to fast waves such as the Whistler waves [3] and increasing requirements for quick numerical experiments, Large Eddy Simulation(LES)-approaches are examined. Adopting the Hall MHD model as an alternative to the MHD equations, some basic properties associated with the LES filtering are studied through benchmark simulations of homogeneous turbulence. We show that some basic quantities in the LES-filtered Hall MHD turbulence are less sensitive than the single-fluid MHD turbulence. We study the reason of the robustness of the Hall MHD system against the LES filtering. We also consider that these results can make sense not only for a helical system but for the other system, too, when the short-waves are important.

[1] H. Miura et al., Fusion Science and Technology Vol.51 (2007) 8.

[2] H. Miura and N. Nakajima, 22th IAEA Fusion Energy Conference (2008) TH-P/9-16.

[3] S. Mahajan and H. Miura, J. Plasma Phys. Vol.75 (2009) 145.

THS/P5-13

Theory for Neoclassical Toroidal Plasma Viscosity in Tokamaks

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Error fields and resistive magnetohydrodynamic (MHD) modes are ubiquitous in real tokamaks. They break the toroidal symmetry in $|B|$ in tokamaks. Here, B is the magnetic field. There are two mechanisms that break the symmetry on the perturbed magnetic surface: one is the perturbed field itself and the other results from the distortion of the magnetic surface due to the perturbed field. The broken toroidal symmetry leads to enhanced neoclassical toroidal plasma viscosity and consequently the rate of the toroidal flow damping. The neoclassical toroidal plasma viscosity also results in a steady state toroidal plasma flow even without toroidal momentum sources, a phenomenon has been observed in stellarators. In addition, the neoclassical toroidal plasma viscosity in the vicinity of the magnetic islands provides a mechanism to determine the island rotation frequency, which is an important quantity for the island stability. All these physics consequences are of interests to fusion grade tokamak devices such as International Thermonuclear Experimental Reactor (ITER). ITER is expected to have low toroidal rotation. Thus, understanding viscosity becomes crucial to predict rotation in ITER.

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THS/P5-14

Analytic Theory of a Matching Problem Generalized for Stability Analysis of Resistive Wall Modes in Rotating Plasmas

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New matching theory is proposed for stability analysis of Resistive Wall Modes (RWMs) in rotating plasmas. It is found that for rotating plasmas, the classical asymptotic matching theory becomes invalid in principle and the matching problem should be generalized to use inner layers with finite width to overcome the difficulty that a resonant surface that must be contained in the layers exists somewhere in a finite range depending on the unknown rotation frequency of the RWM. For the first time, numerical analysis based on the generalized matching problem shows that the rotation in one inner layer significantly affects the stability of RWMs. Analytic theory for the generalized matching problem is developed to physically interpret the results by numerical analysis.

THS/P7 — Poster Session

THS/P7-01

Second Stable Regime of Internal Kink Modes Excited by Barely Passing Energetic Ions in Tokamak Plasmas

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The internal kink (fishbone) modes, driven by barely passing energetic ions (EIs), are numerically studied with the spatial distribution of the EIs taking into account. It is found that the modes with frequencies comparable to the toroidal precession frequencies are excited by resonant interaction with the EIs. Positive and negative density gradient dominating cases, corresponding to off-and near-axis depositions of neutral beam injection (NBI), respectively, are analyzed in detail. The most interesting and important feature of the modes is that there exists a second stable regime in the higher β_h (=pressure of EIs / toroidal magnetic pressure) range, and the modes may be excited by the barely passing EIs in a region of $\beta_1 < \beta < \beta_2$ only. Besides, the unstable modes require minimum density gradients and minimum radial positions of NBI deposition. The physics mechanism for the existence of the second stable regime is discussed. The results may provide a means of reducing or even preventing the loss of NBI energetic ions and increasing the heating efficiency by adjusting the pitch angle and driving the system into the second stabile regime fast enough.

THS/P7-02

Lagrangian Approach to Resonant Three-mode Interaction in Magnetohydrodynamics

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Three-mode coupling due to magnetohydrodynamic (MHD) nonlinearlity is studied for the understanding of saturation mechanism of the toroidal Alfven eigenmode (TAE) that is destabilized by energetic particles. An analytic expression of the coupling coefficient among ideal MHD eigenmodes is derived in a general manner, which quantifies the effectiveness of the nonlinear mode coupling (i.e., the parametric decay). This formulation for the first time enables the analysis of the global mode coupling in real geometry. The well-known equation of the displacement vector field in linear MHD theory is rigorously extended to nonlinear regime by taking the Lagrangian approach. The coupling coefficient is written in terms of the linear displacement vector fields, so that it can be directly and systematically evaluated by utilizing the existing linear stability code.

THS/P8 — Poster Session

THS/P8-01

Geodesic Acoustic Modes in Rotating Large Aspect Ratio Tokamak Plasmas

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Analytical theory of Geodesic Acoustic Modes (GAM's) is modified for a general case of rotating tokamak plasma. Both toroidal and poloidal components of steady-state plasma rotation are taken into account. For large aspect ratio tokamaks, the dispersion relation of electrostatic perturbations is derived analytically in the frame of one-fluid ideal magnetohydrodynamics. In the case of small (compared to the sound frequency) angular rotation velocity, two solutions of dispersion relation are found. The first one is the standard GAM modified by the rotation effects. The second mode has a frequency close to the frequency of acoustic mode. The new GAM is induced by poloidal plasma rotation. This mode appears as a consequence of the Doppler frequency shift in the side-band components of plasma density, pressure and parallel velocity perturbations. The side-bands arise as the curvature driven response to the electrostatic potential perturbation with $m = 0$ (m is the poloidal wavenumber). The Doppler frequency shift is caused by poloidal rotation and has opposite signs for the $m = 1$ and $m = -1$ side-bands. Unlike the case of tokamak equilibrium with isothermal magnetic flux surfaces, no new low-frequency GAM arises in the case of purely toroidal plasma rotation in tokamak with isentropic magnetic surfaces. The pure toroidal flow results only in the up-shift of GAM frequency.

THS/P8-02

Turbulent Generation of Flows and Magnetic Field at the Rational Magnetic Surfaces of a Tokamak

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Comparative analysis of generation of large-scale structures, zonal flows and streamers, by drift wave turbulence is conducted for periodic systems with magnetic shear such as a tokamak. In a strong magnetic field dynamics of quasi two-dimensional perturbations strongly depends on the value of the wave vector along the magnetic field. When the parallel wave vector is significantly large, so that the parallel phase velocity of perturbation is small compared to electron thermal velocity, the parallel electron motion results in a finite electron density perturbation. It follows the Boltzmann distribution. However, for large-scale structures with poloidal and toroidal symmetry $m = n = 0$, and the parallel wave vector is zero. This results in strong reduction of density perturbation for $m = n = 0$. This difference has profound consequences for generation of large-scale zonal flows and streamers due to different structure of the nonlinear interaction matrix. The interaction term has a structure similar to the standard convective nonlinearity for zonal flows, while for streamers it has the structure of the Hasegawa-Mima nonlinearity (which is the higher order due to a small parameter assocaited with a finite ion Larmor radius). Respectively, zonal flows have the larger growth rate gamma(ZF) compared to that of the streamers. It is shown that 3D electromagnetic helical perturbations will have the growth rate comparable to that of zonal flows if their symmetry coincides with the symmetry of rational magnetic surface, $m = nq$. The field line bending provides a stabilizing effect and thus determines the radial localization of such structures. Therefore, it is expected that three-dimensional structures of flows and magnetic field will be preferentially generated at the rational magnetic surfaces of a tokamak with a growth rate of order gamma(ZF). This theoretical result may corroborate existing experimental correlations of large-scale shear flow structures with rational magnetic surfaces of a tokamak.

THS/P8-03

Integrated Non-Modal Linear and Renormalized Nonlinear Approach to the Theory of Drift Turbulence in Plasma Shear Flow

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In our report, we present the results of the investigations of the temporal evolution and saturation of drift turbulence in shear flows, which are based on the method of shearing modes, developed by Kelvin, or a so-called non-modal approach. We extend the method of shearing modes, developed originally for the linear fluid analysis of the temporal evolution of the separate spatial Fourier harmonic, onto the development of the renormalized non-linear non-modal hydrodynamic theory of drift turbulence of plasma shear flows, non-modal kinetic theory of plasma shear flows, and renormalized non-linear non-modal kinetic theory of the turbulence of plasma shear flows. The consistent investigation of the temporal evolution of the turbulence in plasma shear flows requires all these theories. The nonlinear integral balance equation for electrostatic potential, which accounts for the random turbulent convection of plasma parcels due to non-modal turbulence, is obtained. Level of drift turbulence is estimated. In contrast to the case of plasma without shear flows, it holds only for a limited time. The time evolution of the potential continues by the non-modal decreasing of its amplitude as the negative second degree of time. Because of the secular growth of the component of the wave number along the velocity shear, the results obtained on the base of fluid equations have a limited validity. For that reason non-modal approach to kinetic theory is developed. We obtain the integral equation for electrostatic potential, in which velocity shear manifests itself as the non-modal time-dependent effect of the finite Larmor radius. The non-modal evolutionary solutions of this equation are obtained for hydrodynamic and kinetic drift-type instabilities of plasma shear flows. The renormalized non-linear non-modal kinetic theory, which accounts for the scattering of ions across the magnetic field due to their interactions with ensemble of shearing modes, is developed

THS/P8-04

Spatiotemporal Chaos, Stochasticity and Transport in Toroidal Magnetic Configurations

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In order to get a more complete insight into the transport processes, zonal flow dynamics and the nature of stochasticity in toroidal systems (helical and tokamak) we analyze nonlinear gyrokinetic Vlasov simulation (GKV) results for the tokamak and the standard and the inward shifted helical configurations from the aspect of nonlinear dynamic systems theory.

The basic formulae for describing the drift wave turbulence in magnetically-confined plasmas are given by the gyrokinetic equations, where time-evolution of the one-body distribution function is described as a nonlinear partial differential equation defined on the five-dimensional phase space. In the gyrokinetics, the finite gyro-radius effect is introduced while the gyro-phase averaging eliminates the fast time-scale phenomena associated with gyro-motions. The nonlinear gyrokinetic equation of the perturbed gyro-center distribution function in the low-beta (electrostatic) limit, is numerically solved in the GKV code. Effects of the helical confinement field are introduced through variation of the magnetic field strength **B** along the field line.

In both the tokamak and the helical case we show how spatiotemporal chaos (STC) develops and how stochasticity arises from STC. We show a method to distinguish each mode in the spatiotemporal evolution which lead to the formation of zonal fows.

Reduction of transport in the inward shifted configuration with respect to the standard one is interpreted as a consequence of STC suppression while analogous phenomenon is noticed in the purely temporal behavior of plasma dynamics. Hence, in the helical configuration the inward shifted helical configuration represents means for control of spatiotemporal chaos.

Layapunov exponents, evaluated with respect to the complete spatiotemporal dynamics, are shown to be directly related to transport properties. In particular, we show how ion heat conductivity may be evaluated directly from Lyapunov exponents. Finally, new directions in the use of nonlinear dynamics theory as means for understanding confined fusion plasma are discussed.

THS/P8-05

Effects of Trapped Electrons in Collisionless Damping of Geodesic Acoustic Mode

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We use the gyrokinetic toroidal code (GTC) to study the collisionless GAM properties with kinetic electrons. The residual level of the zonal flow agrees well with the theory and is insensitive to the kinetic electrons. In the adiabatic electron simulation in the small ϵ limit, the frequency and damping rate of the GAM correspond well to the analytic results. In the simulation with kinetic electrons, we find that the GAM frequency is insensitive to the trapped electrons while the GAM damping rate is greatly enhanced by the trapped electrons especially in the high-q region. The contribution of passing electrons to the GAM collisionless damping is much smaller than the trapped electrons. The resonance of the trapped electron bounce motion with the GAM oscillation is clearly seen from the perturbed electron distribution function in the $E - \mu$ phase space (E and μ are the electron energy and magnetic moment, respectively).

THS/P8-06

Zonal Flow Modes in Tokamak Plasma with a Dominantly Poloidal Flow

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It is possible that a strong poloidal plasma flow exists in tokamak plasmas due to strong radial electric field, Stringer spin-up effect and external momentum input. In this work, the axisymmetric electrostatic oscillations (Zonal flow modes) are investigated under such a situation. The flow velocity takes the form under an ideal MHD limit, with the poloidal and the toroidal flow in the same order and comparable with the poloidal sound speed. It is found that the frequencies of the two branches, corresponding to the normal GAMs and the sound waves (SWs) in static plasmas, increase significantly with the poloidal Mach number. The standing wave form of the density perturbation in static plasmas is also changed and superimposed with a small amplitude traveling wave for both GAMs and SWs.

THW — Oral

(*Magnetic Confinement Theory and Modelling: Wave-plasma interactions -current drive, heating, energetic particles*)

THW/2-1

Low Threshold Parametric Decay Instabilities in Tokamak ECRH Experiments

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ECRH at power level of up to 1 MW in a single microwave beam is often used in present day tokamak and sellarator experiments and planed for application in ITER for neoclassical tearing mode island control. According to theoretical analysis of parametric decay instabilities (PDI) thresholds, the typical RF power at which these nonlinear effects can be excited at tokamak plasma parameters is around 1 GW. Therefore it is taken for granted that wave propagation and absorption in ECRH experiments is well described by linear theory and thus predictable in detail.

However last year the first observations of the backscattering signal in the $200 - 600$ kW level second harmonic ECRH experiment at Textor tokamak were reported. This signal down shifted in frequency by approximately 1 GHz, which is close to the lower hybrid or ion Bernstein (IB) wave frequency under the Textor conditions, was strongly modulated in amplitude at the $m = 2$ magnetic island frequency.

In the present paper the experimental conditions leading to substantial reduction of backscattering decay instability threshold in tokamak ECRH experiments are found. The parametric decay of the pump X-mode into backscattering X-mode and IB wave is considered accounting for poloidal magnetic field inhomogeneity and experimental non monotonic density profile in the vicinity of magnetic island O-point. It is shown that magnetic island at the low field side of the torus serves as 2D waveguide for the daughter IB wave. The drastic (four orders of magnitude) decrease of the PDI threshold compared to predictions of the standard theory is demonstrated and explained by complete suppression of IB wave convective losses in radial and magnetic field direction and their substantial reduction in the third direction. The role of parametrically driven IB wave absorption in ion acceleration often observed in ECRH experiments is discussed.

THW/2-2Ra

Damping, Drive and Non-Linear Effects of Kinetic Low-Frequency Modes in Tokamaks

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In recent years, rather global modes well below the TAE frequency have been identified by internal fluctuation measurements such as soft X-ray and reflectometry measurements. At ASDEX Upgrade these BAEs and BAAEs are often located at or close to the $q = 1$ surface, that is typically relatively core localized. In the presence of on-axis ICRF heating steep gradients of both energetic particles and background ions in this core region can drive these otherwise strongly ion-Landau-damped modes unstable. Whereas previously only one BAE at around 80 kHz was seen, dedicated experiments revealed recently that several BAEs with different poloidal mode numbers determined by magnetic and SXR measurements can be excited simultaneously, as seen also at Tore Supra and predicted by kinetic dispersion relations. An extended version of a kinetic dispersion relation for the Alfven-Acoustic continuum including the diamagnetic effects due to the background is applied for determining the continuum gap structure in the plasma core. The comparison to experimental data shows that this fully kinetic treatment is crucial for understanding the physics of these low frequency modes. For the fast particle drive, the existence criteria and the mode structures, the linear gyrokinetic code LIGKA is used, allowing for a detailed investigation of the transition from the Alfvenic into the KBM regime with increasing mode numbers.

The importance of these core localised modes becomes obvious when the overall transport of energetic particles is investigated. It is well known that several modes with similar frequency at different radial locations cause an increased transport due to resonance overlapping in phase space. The question addressed here is how modes with very different frequencies and resonance conditions can interact, e.g. TAEs and BAEs, as seen experimentally. It is found that a special class of particles that is resonant with both modes via different resonance conditions causes the amplitudes of both modes to couple. Therefore, the orbit width of these particles rather than the radial mode overlap is the crucial parameter for effective mode coupling.

Furthermore, an extended version of the HAGIS code across the separatrix including finite orbit width effects allows for a detailed modeling of fast ion losses as measured by the fast ion loss detector at ASDEX Upgrade.

THW/2-2Rb

Simulations of Energetic Particle-driven Instabilities with Source and Sink

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The energetic particle-driven instabilities and energetic particle transport are important for burning plasmas such as ITER. Here we report new results of nonlinear simulations of energetic particle-driven Toroidal Alfven Eigenmodes(TAE) with particle collision, particle source and sink, as well as with plasma micro-turbulence-induced energetic particle radial diffusion. The results of TAE saturation level scaling with collisional frequency agree very well with analytic theory. The simulation results also show that the effects of plasma micro-turbulence-induced radial diffusion might be more important than the collisional effects for determining thesaturation level of energetic particle-driven Alfven modes in the present day tokamaks and future reactors. Finally, the simulation of beam-driven TAEs in the National Spherical Torus Experiment(NSTX) reveals the importance of plasma toroidal rotation on mode structure and the significant effects of wave particle resonance overlap on the mode saturation.

THW/2-3Ra

Simulation Study of Nonlinear Magnetohydrodynamic Effects on Alfvén Eigenmode Evolution and Zonal Flow Generation

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Nonlinear magnetohydrodymamic (MHD) effects on Alfven eigenmode evolution were investigated via hybrid simulations of an MHD fluid interacting with energetic particles [1]. The investigation focused on the evolution of an $n = 4$ toroidal Alfven eigenmode (TAE) which is destabilized by energetic particles in a tokamak. In addition to fully nonlinear MEGA code [2], a linear-MHD version of MEGA code was used for comparison. The only nonlinearity in that linear code is from the energetic particle dynamics. No significant difference was found in the results of the two codes for low saturation levels, $\delta B/B \sim 10^{-3}$. In contrast, when the TAE saturation level predicted by the linear code is $\delta B/B \sim 10^{-2}$, the saturation amplitude in the fully nonlinear simulation was reduced by a factor of 2 due to the generation of zonal ($n = 0$) and higher-n ($n \geq 8$) modes. This reduction is attributed to the increased dissipation arising from the non-linearly generated modes. The fully nonlinear simulations also show that geodesic acoustic mode is excited by the MHD nonlinearity after the TAE mode saturation.

[1] Y. Todo, H.L. Berk, and B.N. Breizman, to appear in Nucl. Fusion 50 (2010).

[2] Y. Todo and T. Sato, Phys. Plasmas 5, 1321 (1998).

THW/2-4Ra

Kinetic Thermal Ions Effects on Alfvenic Fluctuations in Tokamak Plasmas

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The early observation of beta induced Alfvén eigenmodes (BAE) and a variety of recent experimental observations have attracted attention on studying the low-frequency Alfvénic fluctuations in tokamaks. The generalized fishbone-like dispersion relation theoretical framework has been adopted for extending the hybrid model by taking into account both thermal ion compressibility and diamagnetic effects in addition to energetic particles(EP) kinetic behaviours. The extended model has been used for implementing an eXtended version of HMGC (XHMGC). In general, the new version of HMGC can have two species of kinetic particles. On one hand, one can use XHMGC for investigating thermal ion kinetic effects on Alfvénic modes driven by EP. In this case, EP dynamics contribute in the ideal MHD region; while wave-particle resonances with core-plasma ions are important only in a narrow inertial layer centred about the mode rational surface, where the dynamics of EP can be neglected due to their large perpendicular orbits (compared to the layer width). On the other hand, it may be interesting to use XHMGC as a tool to simulate two coexisting EP species, generated e.g. by both ICRH and NBI heating, in order to study linear excitation of Alfvénic fluctuations and Energetic Particle Modes (EPM), as well as the interplay between the respective nonlinear physics. Results of initial-value simulations show that the observed frequency is always slightly higher than the BAE accumulation point and is the same at different radial positions; consistent with the characteristics of a discrete BAE-SAW eigenmode (termed as kinetic BAE or KBAE); however, no discrete eigenmode is found within the gap when MHD is ideally stable. Meanwhile, preliminary simulations of KBAE/EPM driven by purely circulating EP have also been done. So far, the results show that the mode frequency is higher than either theoretical BAE accumulation point frequency or EP transit frequency, and increases with thermal ion temperatures. Comparing numerical results with theoretical predictions with EP physics suggests deriving δW_k including finite $k_{\|}$ effects on EP wave-particle resonances. Once the linear mode structures of KBAE are assessed and comparisons with analytic theory are done, nonlinear simulation will be carried out with thermal ion kinetic effects and obliquely injected EP.

THW/P2 — Poster Session

THW/P2-01

Kinetic Integrated Modeling of Heating and Current Drive in Tokamak Plasmas

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In order to self-consistently describe heating and current drive and various influences of energetic particles in tokamak plasmas, we have been developing a kinetic integrated modeling code TASK3G. This modeling is based on the behavior of the momentum distribution function of each particle species. The time evolution of the momentum distribution function is described by a newly-extended Fokker-Planck component TASK/FP and the influence of energetic particles on global stability is studied by a full wave component TASK/WM. Self-consistent analysis of multi-scheme heating in a ITER plasma is demonstrated for the first time including radial transport and fusion reaction rate calculated from the momentum distribution functions. We have introduced several kinds of turbulent diffusion coefficient and inward pinch; with and without energy dependence, fixed radial profile, and turbulent transport models used in transport simulations. The simulation results are compared with experimentally observed radial profiles of temperature and energetic particles. The full wave analysis including the effects of energetic particles has indicated their influence on the power deposition profiles in ICRF heating. The effects of energetic particles on the linear stability of Alfven eigen modes and low-frequency global eigen modes are also studied.

THW/P2-02

Full Tokamak Discharge Simulation for ITER

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A full tokamak discharge simulator has been developed by combining CRONOS and DINA-CH. The two codes have been coupled in a modular way using an explicit data exchange scheme and used to study the proposed ITER operation scenarios. Firstly, the inductive 15MA ITER H-mode scenario has been simulated as a demonstration of the capability of the simulator, as well as a design study in itself. The engineering and operational constraints are taken into account for whole operation phases including the plasma current ramp-up, flat-top and ramp-down. The poloidal field (PF) coil current limit, poloidal flux consumption, plasma shape evolution and vertical instability associated with a high internal inductance were investigated to access the feasibility of the scenario. Secondly, lower hybrid (LH) assisted plasma current ramp-up has been studied. Application of the LH during the current ramp-up was effective in lowering the internal inductance to be favourable for vertical stabilization of the plasma and also in reducing the poloidal flux consumption. A slightly reversed or flat target safety factor profile required to operating ITER plasmas in advanced tokamak regimes was achieved avoiding the onset of sawteeth during the current ramp-up phase. Lastly, ITER hybrid mode operation scenario has been studied aiming at achieving a stationary flat safety factor profile at the beginning of the current flat-top phase. The plasma current ramp-up scenario is generated by tailoring the initial part of the inductive 15MA H-mode ITER operation and the effect of different heating and current drive schemes on the evolution of the safety factor profile has been investigated. Application of a near on-axis electron cyclotron current drive (ECCD) appears to be effective in modifying the safety factor profile compared to the far off-axis lower hybrid current drive (LHCD), at least on short time scales.

THW/P2-03

Potential of the ICRF Heating Schemes foreseen for ITER's Half-Field Hydrogen Phase

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[11]*See Appendix of F. Romanelli et al., Nucl. Fusion* **49** *(2009) 104006.*

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In view of enhancing the knowledge of the ion cyclotron resonance frequency (ICRF) heating schemes in the non-activated phase of ITER, a detailed assessment of the fundamental H and the 2nd harmonic ^{3}He ICRF heating scenarios in majority H plasmas doped with modest up to large concentrations of ^{3}He was undertaken. The numerical studies were performed using the 1D TOMCAT [1] code and the 2D CYRANO [2] code adopting the equilibrium profiles computed for the initial operation phase of ITER. The nominal ITER half-field of 2.65 T and the ICRF frequencies for those heating scenarios (42 MHz for fundamental H and 52MHz for 2nd harmonic ^{3}He heating) were closely reproduced in dedicated experiments performed in the JET tokamak.

Simulations show that fast-wave electron heating will dominate in both the fundamental H majority and the 2nd harmonic ^{3}He ICRF heating schemes, particularly at the rather modest densities and temperatures expected in the initial operation of ITER. The single-pass absorption of these scenarios is generally low because of the incorrect polarization of the RF fields in the fundamental H ICRF heating case and because of the relatively low minority concentrations expected in the ITER plasmas for the 2nd harmonic ^{3}He heating case. Increased plasma density and temperature both help to enhance the single-pass absorption but in general electron rather than ion damping is primarily improved. For the fundamental H ICRF heating case, the ion temperature plays a major role on the scenario's performance but bulk ion temperatures of about 25 keV are necessary to obtain dominant ion heating (with a total single-pass absorption of $\sim$ 60%). For the 2nd harmonic ^{3}He ICRF heating case, the ion temperature also influences but the ^{3}He concentration is the main actuator on the performance of the heating scheme. However,

the simulations suggest that rather high concentrations (X[^{3}He]$\sim$ 25%) would be needed to achieve prevailing ion heating in this case, with a total single-pass absorption of $\sim$ 40%. The single-pass absorption is only about 25% for the typical X[^{3}He]=4% value proposed for ITER.

[1] D. Van Eester et al., Plasma Phys. Control. Fusion 40 (1998) 1949-1975.

[2] P.U. Lamalle (1994) PhD Thesis LPP-ERM/KMS Report 101 Université de Mons, Belgium.

THW/P2-04

Modeling of Nonlinear Electron Cyclotron Heating during ECH-assisted Plasma Startup in a Tokamak

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In this research, we investigate ECRH for low-energy electrons analytically and numerically, which is relevant to ECRH-assisted plasma start-up in a tokamak. From the previous experimental studies, it is well-known that the first harmonic ECRH is more effective than the second harmonic ECRH. In this work, we developed an analytic model of ECRH in the start-up process and present comparisons of the efficiency between the first harmonic and the second harmonic. It is found that electrons gain a large energy from the waves at first harmonic resonance up to several hundred eV. However, electrons gain a large energy from the waves only up to $\sim$eV and energy-gain starts decreasing afterward at second harmonic resonance. When seed electrons are heated from the room temperature to far away above the ionization energy, seed electrons can bring about an avalanche of electrons. Thus, pre-ionization with the second harmonic can be delayed since electrons need more time to be heated up to the breakdown temperature due to the slow heating speed compared to the first harmonic ECRH.

THW/P3 — Poster Session

THW/P3-01

Fast Ion Power Loads on ITER First Wall Structures in the Presence of NTMs and Microturbulence

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We have already studied the wall loads caused by fast ions in a variety of ITER plasmas using the 5D Monte Carlo guiding-center code ASCOT. The simulations were performed for different magnetic configurations including FIs and none or more TBMs.

In earlier simulations, the fast ion transport was assumed purely neoclassical. In MHD quiescent plasmas this was believed to be true until the first NBI current drive experiments with the AUG tangential beams failed to demonstrate the predicted levels of off-axis current. The subsequent theoretical work has revealed that microturbulence can induce additional transport even for the fast ions. Furthermore, it is highly unlikely that the ITER plasmas will be MHD quiescent: the massive fast ion population consisting of the fusion alphas with energies up to 3.5 MeV drive a multitude of energetic particle modes. Furthermore, ITER is prone to NTMs with substantial island structures. All these MHD phenomena can contribute to increased transport of fast ions in the core plasma.

We are incorporating non-neoclassical effects into ASCOT. Here we present the first simulation results of fast ion power loads to ITER plasma facing components where realistic 3D magnetic field and most recent 3D wall structure are used, and the fast ion redistribution due to microturbulence and NTMs is included.

The simulations are carried out for ITER Scenario 2 (standard H-mode) and Scenario 4 (steady-state operation). In the neoclassical study, the wall power loads were found insignificant in Scenario 4 and much larger but still easily tolerable in Scenario 2. This is due to the different plasma profiles: in Scenario 4, thermal fusions and ionization of NBI particles occur closer to the plasma core. Thus the redistribution of fast ions due to non-neoclassical effects can be particularly treacherous in Scenario 4: drift islands and anomalous transport lead energetic ions to the edge where ripple and TBM induced transport processes rapidly take them to the wall.

Preliminary results indicate that when ripple is not taken into account both the islands and anomalous diffusion have little effect on the wall loads. However, together with ripple, large islands increase the wall loads significantly. Luckily most of the load is still found on the limiters. The effect of anomalous diffusion on the wall loads is small compared to that of the islands.

THW/P4 — Poster Session

THW/P4-01

Gyrokinetic Simulations of Energetic Particle Driven TAE/EPM Transport Embedded in ITG/TEM Turbulence

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Energetic particle transport from local high-n toroidal Alfvén eigenmodes (TAEs) and energetic particle modes (EPMs) has been simulated with the gyrokinetic code GYRO [1]. Linear and nonlinear simulations have identified a parameter range where the TAE and EPM are unstable alongside the well-known ion-temperature-gradient (ITG) and trapped-electron-mode (TEM) instabilities. A new eigenvalue solver in GYRO facilitates this mode identification. States of nonlinearly saturated local TAE/EPM turbulent intensity are identified, showing a "soft" transport threshold for enhanced energetic particle transport against the TAE/EPM drive strength parameter n_{EP}/n_e. The very long-wavelength TAE/EPM transport is likely saturated by nonlinear interaction with ITG/TEM-driven zonal flows. These states are accessible over a relatively narrow range of TAE/EPM drive strength. Within this range, and in the local limit employed, TAE/EPM driven transport more closely resembles driftwave microturbulent transport than "stiff" clamped critical pressure gradient ideal MHD transport.

[1] J. Candy and R.E. Waltz, Phys. Rev. Lett. **91**, 045001 (2003)

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THW/P4-02

Flow Generation Associated with RF Current Drive in a Tokamak Plasma

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Radial electric field can be built by the charge accumulation due to resonant trapped electron pinch, when the rf wave is injected, and then drive the toroidal flow. The resonant pinch effect is evaluated for the parallel-drive scheme. It is found the radial electron flux and its driven electric field is proportional to the rf power density and also proportional to the rf driven current density. However, the efficiency of the radial electron flux drive decays exponentially with respect to the square of the parallel phase velocity of the wave. For LHW, the parallel phase velocity is a few times of the electron thermal speed, therefore the momentum deposited on trapped electron is rare otherwise the upshift of the $N_{\|}$ spectrum is obvious. To generate typical radial electric field with order of 10 kV/m, it is needed ~ 10 kW/m^3 power density deposited on the thermal electrons, which is a few percentage of 1 MW/m^3 for current drive. When low frequency wave, e.g. fast wave or Alfven wave is applied, the current drive efficiency is weaken due to enhanced collisional dragging and electron trapping, but there is high efficiency for the electric field generation and flow drive.

THW/P4-03

Simulation Study of Toroidal Flow Generation by the ICRF Minority Heating

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The toroidal flow generation by the ICRF heating is investigated in the tokamak plasma applying GNET code, in which the drift kinetic equation is solved in 5D phase-space. In this study we assume a tokamak plasma similar to the Alcator C-mod plasma as a first step. We perform the simulation using GNET code until we obtain a steady state distribution of energetic minority ions. The obtained distribution of energetic minority ions is analyzed and the toroidal flow is evaluated integrating the distribution function with the weight of the local toroidal velocity. It is found that a co-directional toroidal flow is generated outside of the RF wave power absorption region. The dominant part of toroidal flow does not depend on the sign of $k_\parallel$ and the simulation result with the symmetric $k_\parallel$ heating also show a toroidal flow similar to the $k_\parallel$ independent part. Futhermore when we change the sign of the toroidal current we obtain a reversal of the toroidal flow velocity, which is consistent with the experimental observations.

Two models to explain the obtained toroidal flow are studied. We, first, consider the torque driven by the ICRF heating based on the gyrokinetic theory including the wave-particle interaction. We derived the net torque due to the ICRF heating and found that the ICRF driven torque is related the toroidal magnetic drift of accelerated energetic ions. It is found that there exists a contribution by the toroidal magnetic drift but the strength is one order small compared with the simulated one. Next we consider the orbit of energetic tail ion accelerated by the ICRF heating. The energetic trapped ions exist on the different minor radius during one banana bounce motion due to the finite banana size. If there is a magnetic shear there exists a net toroidal motion during one bounce. Additionally, the poloidal magnetic drift increases the toroidal direction motion in the outer side of banana orbit and decreases in the inner side. This also enhances the net toroidal motion. We can see the energy dependent toroidal direction motion. No clear net toroidal motion can be seen in the case no magnetic shear and no poloidal magnetic drift. Also, this motion changes the direction if the toroidal current is reversed. This would be a mechanism to drive the net toroidal flow by the ICRF heating.

THW/P7 — Poster Session

THW/P7-01

Validation of Simulation Capability for RF Wave Propagation and Absorption in the Ion Cyclotron Range of Frequencies on Alcator C-Mod

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In order to optimize the application of RF power in the ion cyclotron range of frequencies (ICRF) for heating, current profile modification, and plasma rotation on Alcator C-Mod, as well as on other present day devices and ITER we have undertaken to verify and validate a predictive simulation capability for propagation, absorption, and mode conversion of ICRF waves. A simulation model has been developed which combines the spectral full-wave solver AORSA with a bounce averaged, zero orbit width Fokker Planck code CQL3D. The AORSA-CQL3D iteration has been carried out in a time dependent calculation in which the evolution of the minority tail energy in the simulation has been compared with the count rate measured by a Compact Neutral Particle Analyzer (CNPA) diagnostic on C-Mod. It is found that the rise time of the energetic ion tail matches the rise time of the CNPA signal quite well, whereas the decay time of the tail in the simulation lags the measured CNPA signal. The non-thermal ion tail from CQL3D is now being used in a synthetic diagnostic for passive charge exchange and the computed signal will be compared directly with the CNPA measurement in order to delineate regimes of validity of the zero ion orbit width approximation used in CQL3D. An important aspect of code verification is the comparison of more complete physics models with so-called "reduced" models. The predictions of wave propagation and absorption from the AORSA solver have been compared with the predictions from a finite ion Larmor radius solver TORIC. Although AORSA is valid for all ion cyclotron harmonics and arbitrary values of ion Larmor radius relative to the perpendicular wavelength, the resulting dense matrix solve is expensive and time consuming. The TORIC solver only retains up to the second cyclotron harmonic, assumes small Larmor radius relative to the wavelength, and treats mode conversion to IBW and ICW with a correction to the conductivity operator. The reduced model (TORIC) was found to agree quite well with AORSA for a D(He-3) plasma in which the He-3 concentration was scanned from the minority heating regime

to the mode conversion regime. Although this reduced model must still be solved on a parallel computing platform, the processor and memory demand is far less, making 3D field reconstructions possible within time dependent simulations.

THW/P7-02

Spontaneous Formation and Evolution of Nonlinear Energetic Particle Modes with Time-Dependent Frequencies

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The near-threshold regimes of wave excitation by energetic particles reveal a rich family of nonlinear scenarios ranging from benign mode saturation to explosive behavior. The choice between these scenarios depends on relaxation processes that restore the unstable distribution function. Our recent analysis shows that only the explosive behavior is possible when drag dominates at the wave-particle resonance. As a result, the instability follows a “hard” non-linear scenario in which the saturation level is insensitive to the small initial growth rate. This finding indicates that Alfvénic instabilities driven by neutral beam injection, or by fusion-born alpha-particles with drag-determined distribution functions should be more prone to the “hard” regime than those driven by ion-cyclotron resonance heating (ICRH) with dominant RF quasi-linear diffusion. The experimentally observed differences between the TAE instabilities on JET and MAST support this theoretical notion.

The explosive nonlinear regimes are known to produce phase space holes and clumps. In previous work, description of such structures was limited to the case of small frequency deviations from the bulk plasma eigenfrequency. However, there are many observations of frequency sweeping events in which the change in frequency is comparable to the frequency itself. The need to interpret such dramatic phenomena requires a non-perturbative theoretical formalism, which this new work provides. The underlying idea is that coherent structures represent traveling waves in fast-particle phase space. A rigorous solution of this type is obtained for a simple one-dimensional model. This model captures the essential features of resonant particles in more general multidimensional problems. The presented solution suggests an efficient approach to quantitative modeling of actual experiments. An effort is currently underway to develop such a numerical procedure.

THW/P7-03

Simulations of NBI Fast Ions in Stellarators

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NBI injectors are widely used in fusion devices since they are an efficient way for plasma heating and fuelling. The ions created by ionization of neutrals have energies much larger than the plasma thermal energy, so they carry a significant fraction of the total energy and current. The understanding of the physics of these fast ions is very important for the design and performance of fusion devices. NBI can be a key tool for plasma heating and fuelling in a stellarator reactor and its current drive capability can be a very useful tool for reaching steady state in tokamaks, making mandatory then estimation of the fast ion distribution function evolution. In the present work the distribution function of fast ions in stellerators is calculated numerically. The orbit code ISDEP is used to evaluate the distribution function statistically using a Monte Carlo method. ISDEP integrates trajectories of test particles (fast ions) that interact with a static background, taking into account guiding center dynamics as well as collisions with thermal ions and electrons. Then the steady state test particle distribution is evaluated using the Green function formalism. ISDEP also provides the profiles of toroidal and poloidal rotation so the plasma current can be computed in arbitrary 3D geometries. The study is performed in two stellarators: LHD and TJ-II, comparing fast ion transport parameters in both devices. Stellarators are suitable platforms to evaluate the current drive capability of NBI, since those devices do not have inductive current and it is easier to identify the external current driven by any method and to validate the calculations. So the study of NBI driven current in stellarators is relevant for ITER and other tokamaks.

THW/P7-04

Energetic Particle Physics in FAST H-Mode Scenario with Combined NNBI and ICRH

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In the Fusion Advanced Studies Torus (FAST), the extreme H-mode scenario requires 40 MW of external heating, mainly supplied by NNBI (10 MW) and ICRH (30 MW). The extreme H-mode is characterized by high magnetic field $B = 8,5$ T and high plasma current $I_p = 8$ MA for a discharge time duration of about 12s, with peak density 5×10^{20} m^{-3} and temperature 9 keV at the plasma centre. Strongly supra-thermal fast ions, such as those expected to be generated in FAST by NNBI and minority ICRH, both in the MeV range of energy, are characterized by small orbit to machine size ratios and predominantly transfer their energy to plasma electrons via collisional slowing down. This energetic ion population can excite meso-scale fluctuations with the same characteristics of those expected in reactor relevant conditions and, for this reason, FAST can address a number of important burning plasma physics issues, such as radial transport of energetic ions due to collective mode excitations, coupling of meso-scale fluctuations with micro-turbulence, etc. Moreover, the combination of ICRH+NNBI in FAST adds great flexibility in the experimental study of these phenomena, for it allows the generation of fast ion populations with different velocity space anisotropy and radial profile; especially power density radial profiles regulate fluctuation intensity profiles and, ultimately, transport processes of both thermal and supra-thermal plasma components. Numerical simulation and modeling are based on the use of various transport codes that are iteratively coupled with a bi-dimensional full wave-quasi-linear solver for ICRH, which also includes the solution of the NNBI-plasma Fokker-Planck equation. Numerical results are obtained self-consistently by the transport code CRONOS with combined ICRH/NNBI heating in the FAST plasma, and ICRH in the frequency range $80 - 85$ MHz, on $1 - 3\%$ He^3 minority concentration in D plasma and 1 MeV energy Deuterium N-beam. The energetic particle distribution functions in the "flat-top" phase can be used as initial condition for a numerical simulation study, investigating the destabilization and saturation of fast ion driven Alfvénic modes by means of a recently extended version of the HMGC code, which is able to simultaneously handle two generic initial particle distribution functions in the space of particle constants of motion.

THW/P7-05

Verification of Gyrokinetic Particle Simulation of Alfven Eigenmodes excited by External Antenna and by Fast Ions

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It is crucial to include kinetic effects of thermal ions and trapped electrons in order to properly account for the damping and, thereby, the instability thresholds for Alfven eigenmodes and energetic particle mode. Therefore, the simulation code must have the capability to resolve the micro-scale thermal ion Larmor radius ρ_i and finite parallel electric field, i.e., a paradigm shift to the global gyrokinetic simulation is necessary for burning plasmas. However, a doubt has been raised whether it is feasible to use the fully kinetic approach for simulating magnetohydrodynamic modes. Here we report an important step in developing the gyrokinetic capability for the Alfven wave physics in magnetic fusion plasmas: verification of the global gyrokinetic toroidal code (GTC) simulations of the toroidal Alfven eigenmode (TAE), reversed shear Alfven eigenmode (RSAE), and ideal ballooning mode. A novel approach is that GTC utilizes an external antenna to excite a damped eigenmode for accurately measuring the real frequency and damping rate (as in recent JET experiments) and the mode structure, which provides a reliable verification of the GTC simulation of fast ion excitation of TAE and RSAE. GTC gyrokinetic simulation is also benchmarked with established hybrid-MHD codes HMGC and TAEFL by suppressing the kinetic effects of thermal ions.

THW/P7-06

Finite Orbit Width Monte-Carlo Simulation of Ion Cyclotron Resonance Frequency Heating Scenarios in DIII-D, NSTX, KSTAR and ITER

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Ion cyclotron resonance frequency (ICRF) wave is one of the main auxiliary plasma heating methods in present tokamak experiments and future ITER. Finite-orbit effects due to drift motion of non-Maxwellian species, such as fast-ion radial diffusion and fast-ion loss to the wall, may significantly modify the ICRF wave propagation and absorption in the plasma. Our latest simulations of high harmonic ICRF heating experiments on DIII-D and NSTX using the 5-D finite-orbit Monte-Carlo code ORBIT-RF self-consistently coupled with 2D full wave code AORSA, including quasilinear and collisional orbit diffusion, validated that finite-orbit width effects and iterations between fast-ion velocity distribution and ICRF wave fields are important in modeling ICRF heating experiments. Outward radial shifts of ICRF heated fast ions computed by ORBIT-RF/AORSA are qualitatively consistent with fast ion spatial profiles measured by fast-ion D_α (FIDA) spectroscopy, whereas zero-orbit simulations using the 3D Fokker-Planck code CQL3D coupled with ray-tracing code GENRAY indicated no outward shifts. These outward shifts are due to radial diffusion of ICRF heated fast ions from drift orbit motion. In ITER, thermal minority ions resonating with ICRF wave can be accelerated to some hundred keV due to the proposed high ICRF power (20 MW). In addition, a large population of fast ions can pre-exist in the form of injected neutral beam ions and fusion-born alpha particles. Preliminary simulation using ORBIT-RF coupled with AORSA for an ITER ICRF heating scenario with an initial Maxwellian distribution indicates a radial outward shift of absorption peak compared to linear absorption directly evaluated using the AORSA dielectric tensor. This radial shift is indicative of finite orbit effect, which may produce more significant outward shift as energetic tails increase. Details of the finite orbit effects of fast ions on KSTAR and ITER ICRF heating scenarios at proposed full powers using self-consistent ORBIT-RF/AORSA will be presented.

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THW/P7-07

Alpha Particle Heating and Current Drive in FRCs and Spherical Tokamaks

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A Monte Carlo code is employed to calculate the profiles of the power deposited and the current generated by the alpha particles produced in D-T fusion reactions in high beta plasmas. The code follows the exact trajectories (no gyro-averaging) and includes particle drag and pitch angle scattering. Different equilibria and plasma parameters are considered. In a relatively small Field Reversed Configuration (FRC) reactor the alpha particles can deposit up to 90% of their power in the plasma, with more than 80% of this power going to the electrons, but produce little current. In a ST reactor (ARIES-ST) almost all the power is deposited in the plasma. The number of alpha particles rotating in the same sense as the current is larger than the number of alpha particles rotating in the opposite sense, thus generating a net current.

THW/P7-08

The Influence of Plasma Shaping Effects on the Damping of Toroidal Alfven Eigenmodes

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The dependence of the damping of toroidal Alfven Eigenmodes (TAEs) on various plasma parameters and shapes is an important field of research in view of the prediction and control of fast particle transport in future fusion experiments like ITER and DEMO. Experimentally, the measurement of TAE damping rates by active in-vessel antennas at JET and Alcator C-mod allows for a direct comparison of both frequency and damping rate with theory. However, it is well known that uncertainties in the equilibrium quantities and profiles can influence the damping significantly, since the damping is exponentially sensitive to these background parameters. Therefore, parameter scans in well diagnosed

discharges are a very promising way to benchmark and validate the theoretical and numerical models against the experiment.

For this reason dedicated experiments were proposed at the Joint European Torus (JET). On the one hand, a slowly relaxing q-profile changes the gap structure significantly and therefore the relation between core and edge damping. On the other hand, an elongation scan was carried out while keeping the density and the central safety factor value fixed. A series of equilibria was carefully reconstructed that are the basis for both a detailed physics analysis of the effect of q-profile relaxation and shaping on the TAE stability as well as a world-wide code benchmark.

The models range from perturbative MHD codes like CASTOR-K and NOVA-K via a warm dielectric tensor model like LEMan and TASK/WM to gyrofluid (TAEFL) and linear gyrokinetic codes (LIGKA). Also codes designed for stellarator geometries like CAS3D-K and AE3D-K are planning to contribute to this benchmark in the tokamak limit. Secondly, the mode frequency, the damping and the mode structure can be compared. Although the TAE mode frequency is a very robust quantity and therefore easy to identify, several modes of the same toroidal mode number can be present in the same gap requiring also a comparison of the mode structure. Finally, after documenting the code-code comparisons in detail, the limits and caveats of comparing the simulation results to the experiment are discussed. A sensitivity study will be crucial for this point. Also the coupling between the vacuum region and the plasma that is neglected by some codes may account for differences between measured and calculated damping rates.

THW/P7-09

Comparison of Quasi-linear and Exact Ion Cyclotron Resonant Heating Diffusion, with and without Finite Width Ion Orbits

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These studies investigate the validity of ICRF quasilinear(QL) diffusion theory by comparison of coefficients calculated with Lorentz orbits in full-wave fields. In addition, we investigate finite-orbit-width effects of ICRF power absorption. Results are obtained within the context of the C-Mod ICRF experiment. QL theory is examined using the new, parallelized, diffusion coefficient code, DC, which calculates RF diffusion by suitable average of results of direct numerical integration of the Lorentz force equation for ion motion in the combined equilibrium fields and the RF full wave EM fields from the AORSA full-wave code. The overall conclusions are that approximation of the excited RF by a single toroidal mode leads to strong correlation pitch angle modification of the RF diffusion, which thereby modifies self-consistent radial power absorption as calculated with the CQL3D Fokker-Planck code. However, inclusion of a fully toroidal mode spectrum results in most of correlations ceasing to exist. Hence, modeling of ICRF power

absorption using a correlation-less QL theory is reasonably accurate, even with a suitably chosen single toroidal mode. Results are presented for the sMC ("simple Monte Carlo") guiding center orbit code comparing ion effects with and without finite-orbit-width effects turned on, using a single toroidal mode representation of the ICRF fields. For minority H heating in C-Mod, the inclusion of finite ion orbits is shown to give a broader collisional power deposition profile and evidence of rf-induced radial transport. In addition, since sMC utilizes the AORSA diffusion coefficients, up-shift in parallel wave number due to the poloidal field is included allowing the present approach to be extended to high harmonic and beam heating on devices including DIII-D and NSTX.

THW/P7-10

Exact Analytical Solution of Alfvén Waves in Nonuniform Plasmas

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The propagation of Alfvén waves in non-uniform plasmas is described through linear second-order differential equations, governing the total pressure and radial plasma velocity. In general, these two differential equations only admit numerical solutions, whose behavior is very much complicated especially near resonance surfaces which encompass essential degeneracies. It is well-known that most existing analytical methods, including the famous Wentzel-Karmers-Brillouin (WKB) approximation fail near such singularities. In this paper, a power analytical method, which is recently developed and named the Differential Transfer Matrix Method (DTMM), is applied to find a rigorously exact solution to the problem of interest. We also present an approximate solution based on the Airy functions.

THW/P7-11

Estimation of Kinetic Parameters based on Chirping Alfven Eigenmodes

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Alfven Eigenmodes (AEs) with frequency sweeping, or chirping AEs, are analyzed based on the so-called Berk-Breizman (BB) model, which is a generalization of the bump-on-tail problem, where we take into account a collision term that represents particle annihilation and injection processes, and an external wave damping accounting for background dissipative mechanisms. Kinetic parameter regimes with chirping nonlinear solutions are delimited in both supercritical and subcritical regimes. A new quasi-periodic chirping regime is found, and the quasi-period of chirping events depends on the linear growth

rate. Based on these new findings, fundamental kinetic parameters such as the linear drive and the external damping rate are estimated by fitting nonlinear chirping characteristics between the experiment and the BB model. This approach is applied to the AEs on JT-60U, which suggests the existence of marginally unstable AEs.

THW/P7-12

Low-Frequency Global Alfven Eigenmodes in Hybrids with Perpendicular Neutral Beam Injection

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A novel global Alfvén eigenmode (GAE) has been predicted, with frequency well below the minimum of the Alfvén continuum. This GAE exists in the tokamak plasmas with broad low-shear central core and safety factor slightly exceeding unity in this region, and is capable of resonating with precession of the trapped energetic ions. This mode has the dominant numbers $m = n = 1$, but the coupling with the upper toroidal sideband is crucial both for eigenmode formation and its excitation by energetic ions. The mode will be excited as the plasma pressure approaches ideal MHD stability limit, which minimizes the ion Landau damping. The properties of novel GAE are consistent with observations of the low-frequency $n = 1$ mode driven by energetic ions in the "hybrid" discharges with record plasma pressures and perpendicular neutral beam injection on the JT-60U tokamak [1].

[1] N.Oyama, A.Isayama, G. Matsunaga et al., Nucl. Fusion 49, 065026 (2009).

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THW/P7-13

Runaway Electron Drift Orbits in Magnetostatic Perturbed Fields

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One of the most crucial problems facing ITER and similar machines is the occurence of plasma-terminating disruptions. During the disruptions, a beam of runaway electrons can form, which has a potential to damage machine parts. On reactor-sized tokamaks, it is desirable to suppress the runaway beam. One possible intervention method is to apply resonant magnetic perturbations (RMP). The efficiency of the concept has been

experimentally proven on several devices, for example JT-60 and TEXTOR. 3D numerical modelling is required to extend the knowledge about the mechanism of the RMP. The calculations are carried out with the ANTS (plasmA simulatioN with drifT and collisionS) code. The code solves the relativistic, gyro-averaged drift equations for the runaway electrons in 3D fields taking collisions with the background particle distributions (Maxwellian) into account. For the purposes of this study the ANTS code has been extended to include synchrotron radiation. A collision operator, which is valid for both low (thermal) and high (relativistic) energies, has also been derived and implemented. In order to benchmark the code, we first compare our results with a device which has shown runaway suppression experimentally. We use a TEXTOR-like equilibrium. The unperturbed magnetic equilibrium is calculated by VMEC, the magnetic field perturbations are modelled to be like the ones produced by dynamic ergodic divertor (DED) coils at TEXTOR. For the work presented here, the DED coils operated so that the formation of magnetic islands with $n = 2$ toroidal mode number is the most favourable. On the plasma edge, due to the steep q profile, the overlapping of islands generate a large ergodic zone, enhancing radial transport and runaway electron losses. A significant difference between the cases with and without perturbation is the rate and the onset time of the losses: in the perturbed cases the losses start earlier than in the unperturbed case. The onset time depends more sensitively on the DED current than the loss rate. Our simulations indicate that the application of the RMP causes a significant decrease in the runaway population. All these results show good agreement with experimental observations.

THW/P7-14

Collisionless Evolution of Isotropic Alpha-Particle Distribution in a Tokamak

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The density of the noninductive current generated due to collisionless motion of alpha-particles in the tokamak magnetic field is calculated. The analysis is based on fully three-dimensional calculations of charged particle trajectories without simplifying assumptions typical for drift and neoclassical approaches. The current is calculated over the entire cross section of the plasma column, including the magnetic axis. It is shown that the current density is not a function of a magnetic surface and is strongly polarized over the poloidal angle. The current density distribution in the tokamak poloidal cross section is obtained, and the current density as a function of the safety factor profile, the tokamak aspect ratio, and the ratio of the particle Larmor radius on the axis to the tokamak minor radius is determined. It is shown that, when the source of alpha-particles is spatially nonuniform, the current density in the center of the tokamak is nonzero due to asymmetry of the phase-space boundary between trapped and passing particles. The current density scaling in the tokamak center differs from the known approximations for the bootstrap current and is sensitive to the spatial distribution of alpha-particles.

THW/P7-15

3D Full Wave Code Modeling of ECRF Plasma Heating in Tokamaks and ITER at Fundamental and Second Harmonics

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We present recent numerically well resolved 3D ECRF STELEC full wave code modelling results for fundamental and second harmonics scenarios in tokamaks and ITER. STELEC code is now updated and includes also non diagonal wave induced plasma current response terms important for quasi perpendicular O-mode launch. The modelled are both ECH scenarios for now projected in Russia spherical IPHT lithium wall tokamak, operating T-10, DIII-D, AUG, JET and TCV machines. Improved numerical resolution for these middle size tokamaks further solidly confirms previously discovered O- and X- modes strong coupling at fundamental harmonic leading to broadened power deposition profiles, in compare with ray tracing predictions, due to influence of Upper Hybrid Resonance (UHR). For T-10 tokamak we consider O-mode outside launch cases with EC resonance in plasma with UHR usual “moon serp” surface and out off plasma fundamental harmonic EC resonance at High Field Side when UHR surface is in-plasma internally closed one with Electron Bernstein Waves (EBW) being excited inside of it due to mode conversion process. Combined self consistent dynamic O-mode, X-mode and EB waves structure in toroidal plasma is intriguing one and is demonstrated. At this last scenario firstly discovered in WEGA stellarator with quasi perpendicular launch the waves are absorbed at second harmonic – due to relativity effects. O-X-B scenario modelling results for several tokamaks also are given. Implications for ITER are given strongly showing re evaluation need for ECH NTM stabilization concept. In second harmonic scenarios we concentrate on X-mode outside launch at sufficiently large plasma densities, some times close (but lower) to this mode density cut off. Exact solution again shows that simultaneously the O-mode is also excited (with smaller amplitude), presumably due to reflection effects and wave depolarization at the wall. High densities regimes are interesting ones because inside of plasma may be created two power deposition peaks. Examples for DIII-D and JET are given. ITER second harmonic scenario for non active plasma phase at half magnetic field was also explored.

THW/P7-16

Electron Bernstein Driven and Bootstrap Current Estimations in the TJ-II Stellarator

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The control of the total parallel current may lead to the possibility of continuous operation in tokamak plasmas and it can provide access to improved confinement regimes in stellarators, by means of control of the rotational transform profile. In fact one of the main lines of research at the stellarator TJ-II is the relation between confinement and the magnetic configuration, putting emphasis on the rotational transform profile. The two main non-inductive parallel currents in plasma confinement devices are the bootstrap and the ones driven by external means, like radio frequency or NBI. The current drive (CD) systems must be appropriated to work on overdense plasmas, since this could be mandatory in a reactor. Therefore, electron Bernstein waves (EBW), which do not present density cut-off have been considered as CD system for TJ-II. In this work we present calculations of the bootstrap and the EBW currents in the dense plasmas confined in a complex 3D confinement device like the TJ-II stellarator.

The precise calculation of the bootstrap current is a numerical challenge, since the error estimates for computations of this current, specially in the long-mean-free-path (lmfp) regime of stellarators, are very large. This issue is particularly relevant for the lmfp regime of stellarators, particularly for TJ-II, which is characterized by its very complex magnetic configuration. A new code, NEO-MC, has been developed in order to overcome this problem. It combines the standard δf method with an algorithm employing constant particle weights and re-discretizations of the test particle distribution. In this way, it is able to provide, for the first time, calculations of the contribution of the lmfp regime to the bootstrap current of TJ-II with very low error estimates.

For a fast estimation of EBCD, different linear models based on the adjoint approach or Langevin equations techniques have been developed in order to simplify the task of solving the kinetic equation by avoiding the usage of time-consuming Fokker-Planck codes. Among them, recent development has been done in the direction of making these models preserve momentum conservation. This is a requirement that the usual high speed limit models did not fulfil. For the present work, all models developed so far have been coupled to the ray tracing code TRUBA, in order to allow for its comparison.

THW/P7-17

Interpretive Modelling of Neutral Particle Fluxes generated by NBI Ions in JET

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Fluxes of neutral particles in the tens-keV energy range are emitted from JET plasmas and measured by a neutral particle analyzer (NPA) with the line-of-sight perpendicular to the plasma in the mid-plane. The experimental NPA data on the energy spectra of neutral fluxes allow for validation of both the model for beam ions and that for electron donors in tokamak plasmas. Our study aims at interpreting and expounding JET NPA measurements of neutral particle fluxes generated by beam ions. Employing both the 3D-Fokker-Planck code FIDIT for fast ions as well as the Monte-Carlo NPA code for bulk neutrals, we model the energy spectra of deuterium neutral fluxes detected by the NPA in JET in the energy range 5 keV $< E <$ 120 keV. The simulation yields good quantitative agreement with the measurements in ICRH-free plasmas. Time-resolved NPA measurements confer useful information about the effect of MHD activity on fast beam ions. To exemplify that we consider a low I/low B JET discharge with a well pronounced NTM and fishbone activity resulting in the reduction of the DD neutron rate during high beam power operation. Analysis and modeling of NPA measurements in this discharge let us conclude that toroidally trapped fast ions are hardly affected by the modes. Therefore, for explaining the neutron rate observed, the confined circulating ions are expected to be redistributed or lost by MHD modes.

THW/P8 — Poster Session

THW/P8-01

Kinetic Theories of Geodesic Acoustic Mode in Toroidal Plasmas

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Geodesic Acoustic Modes (GAM) are shown to constitute a continuous spectrum due to radial inhomogeneities. The existence of a singular layer causes GAM to mode convert to short wavelength kinetic GAM (KGAM) via finite ion Larmor radii (FLR) and finite guiding-center drift-orbit-width (FOW) effects. The dispersion relation of GAM/KGAM with FLR/FOW as well as parallel electric field contributions is derived to demonstrate the mode conversion to KGAM and propagation in the lower-temperature and/or higher-q region. Corresponding collisionless damping of GAM/KGAM excited in the large q region including higher-order harmonics of transit resonances is investigated, and the analytical expression for the damping rate agrees well with numerical results in its validity regime. Excitation of energetic-particle-induced GAM (EGAM) by velocity space anisotropy is investigated taking into account the coupling to the GAM continuous spectrum in the regime where the energetic particle drift-orbit-width is smaller than the EGAM wavelength and the scale length of GAM continuous spectrum. The response of energetic particles is studied nonperturbatively, and both local and nonlocal EGAM dispersion relations are derived assuming a single pitch-angle slowing-down energetic particle equilibrium distribution function. For a sharply localized energetic particle source, it is shown that the mode is self-trapped where the energetic particle is strongest, with an exponentially small damping due to tunneling coupling to outward propagating KGAM. The dispersion relation of nonlocal EGAM shows that the EP drive must exceed the convective damping due to the tunneling coupling to KGAM; resulting in a finite instability threshold for the global EGAM. The pitch-angle threshold for the local EGAM instability is also shown analytically for the single pitch-angle energetic particle distribution function considered here.

THW/P8-02

Physics of Geodesic Acoustic Modes

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Geodesic Acoustic Modes (GAM) are induced by plasma compressibility in toroidal geometry (geodesic curvature) and, in a simplest form, represent the linear eigen-modes of poloidal plasma rotation. GAMs are also often referred as finite frequency Zonal Flows (ZF) because the primary polarization is toroidally and poloidally symmetric ($m = n = 0$) perturbation of the electrostatic potential. GAM are of major interest to the problem of anomalous transport and plasma confinement; it is expected that they play a role in self-regulation of drift wave turbulence through the shearing effects, energy sink, and possibly radial energy wave transport. GAMs are linearly coupled to drift-waves via toroidal side-bands of plasma pressure, and, similarly to Zonal Flows, can be nonlinearly driven by Reynolds stress due to drift wave fluctuations. GAMs are inherently related to Beta Alfven Eigen-modes (BAE) and multiplicity of modes exhibiting geodesic acoustic frequency scaling have been observed high-temperature tokamak plasmas, particularly, in plasmas with large population of high energy particles. In recent years, the theory of GAM progressed to include a variety of kinetic effects, plasma inhomogeneities, linear destabilization effects due to high energy particles and nonlinear drive due to drift wave turbulence. There have been also significant experimental efforts to clarify the key ingredients of GAM physics. The purpose of this talk is to overview the progress and currents status of GAM theory emphasizing the relation to experimental observations.

THW/P8-03

A Simplified Momentum Conservation Analysis on Transport Reduction induced by Zonal Flow and Turbulent Dissipations

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In the present paper, we demonstrate that the generation of zonal flow is always accompanied by that of turbulent dissipations, i.e., turbulent resistivity and viscosity, based on the momentum conservation of toroidal plasma system. Furthermore, the qualitative analysis showes that the electromagnetic micro-turbulence can produce the significant turbulent resistivity and viscosity at or near rational surface. Finally, we give the heat transport coefficients of ions and electrons, including the effects of the turbulent resistivity and viscosity, i.e., containing those of zonal flow. The present results indicate that the turbulent viscosity makes the heat transport coefficient of ions decrease significantly at or near rational surface while the turbulent resistivity reduces largely that of electrons there.

FTP

Fusion Technology and Power Plant Design

FTP — Oral

FTP/1-1Ra

Demonstration of 500 keV Beam Acceleration on JT-60 Negative-ion-based Neutral Beam Injector

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Hydrogen negative ion beams of 490 keV, 3 A and 510 keV, 1 A have been successfully produced in the JT-60 negative ion source with three acceleration stages. This is the first acceleration of the H- ions up to 500 keV at high-current of > 1 A. These successful productions of the high-energy beams at high current have been achieved by overcoming the most critical issue, i.e., a poor voltage holding of the large negative ion sources with the grids of ~ 2 m^2 for JT-60SA and ITER.

To improve voltage holding capability, the breakdown voltages for the large grids was examined for the first time. It was found that a vacuum insulation distance for the large grids was $6-7$ times longer than that for the small-area grid (0.02 m^2). From this result, the gap lengths between the grids were tuned in the JT-60 negative ion source. The modification of the ion source also realized a significant stabilization of voltage holding and a short conditioning time. These results suggest a practical use of the large negative ion sources in JT-60 SA and ITER.

FTP/1-2Ra

Development in Russia of Megawatt Power Gyrotrons for Fusion

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During last years several new gyrotrons were designed and tested at IAP/GYCOM. Main development efforts were spent for development 170 GHz/1 MW/50%/CW gyrotron for ITER and multi-frequency gyrotrons.

The applied test facility at Kurchatov Institute was upgraded to extend its testing capabilities and to approach them to the ITER specifications. The use of evacuated waveguide assumed for ITER will thoroughly solve a problem of RF arcing. The industrial production prototype of the ITER gyrotron was tested at power 1.02 MW in 200 second pulses, 0.65 MW in 800 second pulses. For 1 MW power regime the gyrotron efficiency is 52%. The pulse energies were limited by overheating of the collector insulator. For the new gyrotron version the modifications have been made in the collector insulator cooling system and they will allow the gyrotron to run at ITER nominal parameters. The last gyrotron version operates in LHe-free magnet.

In 2008/2009 an advanced short-pulse (100 mcs) gyrotron model operating with TE28.12 mode was tested at IAP. The model showed a very robust operation (microwave power up 2 MW with electronic efficiency of 34%) at relatively high electron energies necessary to achieve the high goal power.

A gyrotron capable to operate at several frequencies is very attractive for plasma experiments. Now the main problem in development of multi-frequency gyrotron is a construction of reliable or tuneable window. Now three diamond window concepts are under consideration: Brewster-angle window, window with matched surfaces and double-disc resonant window.

FTP/2-1

Engineering Design Evolution of the JT-60SA Project

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The JT-60SA experiment is one of the three projects to be undertaken in Japan as part of the Broader Approach Agreement, conducted jointly by Europe and Japan, and complementing the construction of ITER in Europe. The mission of JT-60SA is to contribute to the early realization of fusion energy by addressing key physics issues for ITER and

DEMO. It is a fully superconducting tokamak capable of confining break-even equivalent deuterium plasmas. The Tokamak has been designed to pursue full non-inductive steady-state operations at high values of the plasma pressure exceeding the no-wall ideal MHD stability limits. For this purpose, JT-60SA has been designed to realize plasma equilibrium covering relatively high plasma shaping with a low aspect ratio A$\sim$2.5 at a maximum plasma current of $I_p = 5.5$ MA.

In late 2007 the BA Parties, prompted by cost concerns, asked the JT-60SA Team to carry out a re-baselining effort with the purpose to fit in the original budget while aiming to retain the machine mission, performance, and experimental flexibility. Subsequently, along 2008, the Integrated Project Team has undertaken a machine re-optimisation followed by engineering design activities aimed to reduce costs while maintaining the machine radius and plasma current. This effort led the Parties to the approval of the new design in late 2008 and hence procurement activities have commenced.

The paper will describe the new baseline and the process that lead to its optimisation.

FTP/2-2

Prospects for Pilot Plants based on the Tokamak, ST, and Stellarator

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A potentially attractive next-step towards fusion commercialization is a "pilot plant", i.e. a device which produces net electricity as quickly as possible and in as small a facility as possible in a configuration directly scalable to a power plant. The pilot plant approach could accelerate the commercialization of magnetic fusion by both demonstrating net electricity production while also carrying forward a high neutron fluence component testing mission needed to ultimately achieve high availability in fusion systems. This paper will explore three configurations for a pilot plant: the advanced tokamak (AT), spherical tokamak (ST), and compact stellarator (CS). These configurations are considered because: the tokamak presently has the most well-developed physics basis, the ST offers simplified maintenance schemes and reduced size and cost, and the CS offers disruption-free operation with low recirculating power. Several noteworthy trends have been found. First, AT pilot plant scenarios have been identified with field, current, normalized beta and fusion power similar to proposed ITER fully non-inductive scenarios but with reduced H_{98} and in a device 30% smaller in major radius and requiring advanced magnet technology with increased average TF current density. For the ST, fusion power and Q required are roughly 2 times the AT values with similar dependence on thermal conversion efficiency. Not unexpectedly, the ST has the smallest major radius and highest average neutron wall loading of all configurations assessed with wall loading values at or above those previously proposed for nuclear component testing. Finally, for the CS, to minimize device size, increase neutron wall loading, and utilize physics assumptions closest to the tokamak, a quasi-axisymmetric (QAS) design with low aspect ratio = 4.5 is considered. CS scenarios

with fusion power ∼500 MW similar to the AT but with 5 times higher fusion Q and engineering Q up to 2.6 have been identified resulting from elimination of external current drive power. Given the importance of high thermal efficiency blankets, pilot plants should arguably be designed to incorporate progressively more advanced blankets over the course of operation. Improved magnet technology is also critical for all configurations. These and other issues will be discussed.

FTP/2-3Ra

Fusion Nuclear Science Facility before Upgrade to Component Test Facility

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The Fusion Nuclear Science Facility (FNSF) enables research to elucidate and resolve multiple-material synergistic effects in the science of fusion plasma material interaction, power extraction, and tritium transport simultaneously encountering four phases of matter (solid, liquid, gas, plasma) across the nuclear, atomic, nano, meso, and macroscopic size scales, in a fully integrated fusion nuclear environment for continuous durations up to $1-2$ weeks. Such an environment can be provided initially in an ST device with the JET-level plasma conditions ($Q = 0.86$ in Hot-Ion H-Mode) providing 0.25 MW/m^2 in outboard fusion neutron wall loading, and subsequently at twice the JET conditions ($Q = 1.7$) to provide 1 MW/m^2. Conservative high-q and moderate-beta plasma conditions are calculated for the FNSF to minimize plasma-induced disruptions and allow the delivery of the required neutron fluence of 1 MW-yr/m^{2}and duty factor of 10%. Fully modular designs for all the chamber components, including the single-turn toroidal field coil center-post, allow component installation and replacement via remote handling, which is required for the research operations of FNSF. Since the device support structures are hidden behind the chamber components, the FNSF provides a ready upgrade path to the Component Test Facility (CTF), which will require more stringent fusion nuclear and operational capabilities, when and if Demo-capable component designs become available. Details of the physics and engineering calculations for the FNSF will be reported, including the major remaining research needed in solenoid-free plasma initiation, ramp-up, and sustainment; and solutions to the continuous divertor operations with heat fluxes up to 10 MW/m^2.

FTP/2-3Rb

A Fusion Development Facility on the Critical Path to Fusion Energy

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A Fusion Development Facility (FDF) [1] based on the tokamak approach with normal conducting magnetic field coils is presented. FDF is envisioned as a facility with the dual objective of carrying forward advanced tokamak (AT) physics and enabling development of fusion's energy applications.

The FDF tokamak is conceived as a double null plasma with very high elongation and triangularity, predicted to allow good confinement of high plasma pressure. High normalized pressure is required in order to utilize full noninductive, high bootstrap current scenarios to demonstrate continuous operation of a tokamak for periods up to two weeks, a necessary step before DEMO and essential to a blanket development mission. The peak exhaust power flux density estimated for FDF is somewhat lower than that of ITER, because of higher core radiation and stronger tilting of the divertor plates. Divertor radiation can further reduce the heat flux. Besides using conservative AT-physics for its baseline operating modes, FDF is capable of further developing all elements of AT physics, qualifying them for an advanced performance DEMO.

FDF has a goal of producing its own tritium and building a supply to start up DEMO using a blanket. Additionally, in port blanket modules, the development of blankets suitable for both tritium production and hydrogen production can be made. With neutron wall loading at the outer midplane of $1-2$ MW/m^2 and a goal of an annual duty factor of 0.3, FDF can produce fluences of $3-6$ MW-yr/m^2, sufficient to enable irradiation qualification of materials and components for at least the first few years of DEMO operation. Key to FDF's ability to carry out all its science and technology studies is the baseline construction scheme. FDF is conceived with the flexibility and maintainability to allow several changeouts of the main full tritium producing blanket, and precision alignment of the axisymmetric divertor structures, in order to assure uniform heat flux deposition for strong target tilt. FDF would be the next major U.S. facility, running in parallel with ITER.

[1] V.S. Chan, et al., Fusion Sci. Technol. 57, 66 (2010).

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FTP/2-4

Scenario Development for FAST in the View of ITER and DEM0

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It is presently widely accepted that a successful exploitation of ITER and a reliable as well as early design of DEMO will need a strong accompanying program. Along this roadmap, a key role should be played by the so-called "Satellite Experiments": JT-60SA and possibly FAST. The two proposals have been designed to be complementary. Recently three new FAST scenarios have been developed. 1) FAST load assembling has been conceived to accommodate 10 MW of NNBI plus 30 MW of ICRH. This allows producing fast particle populations with different anisotropy and profile localization. Transport simulations confirm zero dimensional estimates and demonstrate the possible achievement of $Q \sim 1.5$. 2) One of the critical points of the FAST proposal is the extensive use of ICRH power with first wall and divertor in full tungsten; since 15 MW of ICRH are sufficient for locally achieving ITER relevant values of Beta-Hot in the FAST H-mode reference scenario, a variant of the reference scenario has been studied, where 15 MW of ICRH have been replaced with 15 MW of ECRH at 170 GHz. 3) Recent experimental results show the necessity of suitable magnetic shear and robust plasma rotation to operate with a reliable ITB. ALCATOR experience shows the possibility of achieving ITB operations at high plasma density even without external momentum injection. Numerical simulation results, have shown the possibility to achieve in FAST the same pressure profiles obtained in ALCATOR, with a relevant intrinsic plasma rotation. To tackle the problem of a possible plasma pollution caused by using ICRH together with a tungsten wall, the code TOPICA has been be used to design an antenna, where power-coupling issues and parallel electric field are reduced. In order to test this concept, an actual test bed experiment has been proposed on ASDEX upgrade. Eventually, in order to study the plasma wall interactions in conditions approaching those of ITER and DEMO, the edge behaviour has been analyzed in greater detail by using the EDGE2D and B2 (plus EIRENE) codes. These investigations confirm the very large wall load (up to 18 MWm^{-2}) and the possibility for FAST to work at very high density with a radiative edge.

FTP/3-1

IFMIF: Status of the Validation Activities and of the Engineering Design Activities

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The Engineering Validation and Engineering Design Activities of the International Fusion Materials Irradiation Facility (IFMIF/EVEDA) is one of the 3 projects of the Broader Approach agreement. The activities conducted are:

– The validation of the main technologies of IFMIF through the design, construction and tests of:

  - An Accelerator Prototype, fully representative of IFMIF low energy (9 MeV) accelerator (125 mA of D^+ beam in CW)
  - A Lithium Test Loop, integrating all elements of IFMIF target facility (same flow of 25 mm thick at 15 m/s) and reduced width (1/3)
  - The High Flux Test Module and its rigs containing the samples to be irradiated in a fission reactor

– The engineering design of IFMIF, with the goal to deliver all information enabling to take all decisions for its construction, operation and decommissioning (CODA)

Validation Activities

– Accelerator Prototype: all systems are now defined and the injector, the radiofrequency quadrupole (RFQ), the RF system are now under fabrication. A full scale mockup of the RFQ has been built to check its tunability. Some key technologies of the superconducting RF linac will enable completing its design, with the goal to start its manufacturing end of 2010. The accelerator Building in Rokkasho is now completed. First tests of the injector are foreseen in Saclay end of 2010.

– EVEDA Lithium Test Loop: its assembly and licensing will be completed in Oarai end of 2010, in order to start the experiments in early 2011. Two target assemblies will be successively tested: an "integral concept" in stainless steel and a more ambitious "bayonet" backplate in reduced activation ferritic martensitic steel. All purification and monitoring techniques will be integrated in this important facility.

– High Flux Test Module: a full scale prototype will be manufactured in order to test its thermo-hydraulic properties in a helium loop (nominal inlet pressure 0.3 MPa at 50°C) now operational in Karlsruhe. A rig is under manufacturing and will be irradiated beginning of 2011 in the BR2 reactor at Mol.

Engineering Design Activities

– Creation of a Specification Working Group, in charge of the definition of IFMIF top level specifications: this group has delivered its report in March 2010,

– Delivery of the "Outline Design Report", summarizing referential design and including evolutions with respect to the CDR, former exhaustive report on IFMIF.

FTP/3-2Ra

Recent Progress in Fusion Technologies under the BA DEMO-R&D in Phase 1 in Japan

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As a part of the Broader Approach (BA) program, R&D on DEMO blanket related materials and tritium technology has been started in 2007 by both the EU and Japan. In accordance with the common interest of the EU and Japan, the followings subjects have been applied: reduced activation ferritic martensitic (RAFM) steels; silicon carbide composites (SiC_f/SiC): advanced tritium breeders and neutron multiplier; and tritium. These R&D studies are mainly carried out in Rokkasho site in Japan. Hence, we defined the period of 2007 to 2009 as Phase 1, and have carried out the design and fabrication of the equipment at Rokkasho. Activities for the licensing of the equipment for tritium and beryllium handling have also been carried out. The equipment is the first and quite unique facility in Japan, where tritium, beta and gamma RI species, and beryllium can simultaneously be used. It has also the first challenge in Japan that researchers in various fields of blanket (for instance, materials and tritium) join R&D program at a facility. A large synergy and combined effect can be expected through the R&D activities at the facility. In addition, a series of preliminary studies for the above R&D subjects has been started with Japanese universities. These activities of Japan in phase 1 are summarized in this paper.

The tritium usage amounts of the equipment per day, per 3 months and per year are 3.7, 185 and 1.35 PBq, respectively. The maximum radioactivity for licensing of other RIs is 1/100 of that of tritium. A conceptual and detailed design of the tritium handling equipments was performed in 2007 and 2008: Glove box and hood; and detritiation systems. A main part of the tritium handling equipment, the detritiation systems, has been constructed in 2009. The concentration of beryllium is limited to be less than 0.002 mg/m^3 in a controlled area. The devices must have a tightly closed structure, and these devices installed in the controlled area must be connected to a ventilation system. The main part of the equipment has thus been constructed in phase 1, so that we move to Phase 2 from 2010. We start a series of formal licensing activities, and fully start a set of experiments with a trace amount of tritium and other RI species by using the equipment at Rokkasho site.

FTP/3-2Rb

Tritium Recovery Experiment from Li Ceramic Breeding Material Irradiated with DT Neutrons

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Tritium recovery experiment from Li ceramic breeding material after the DT neutron irradiation has been carried out for the first time in the world. The DT neutron irradiation experiment has been done at the Fusion Neutronics Source facility in Japan Atomic Energy Agency. A cylindrical bulk with beryllium blocks, which is a neutron multiplier material, was used in order to simulate the neutron spectrum in the Japanese Test Blanket Module design. In the berylliumbulk, a stainless steel rectangular container which contained 65.2 g of natural abundance lithium titanate pebble was inserted at about 203 mm depth point from the beryllium bulk surface. The beryllium bulk was placed 200 mm from the DT neutron source and was irradiated with DT neutrons. After the irradiation, the tritium recovery operation was carried out. The irradiated pebble was heated to 800°C with a heater in the stainless steel assembly, where helium gas with hydrogen gas was flowed. Tritium in the flow gas was oxidized to HTO with the CuO bed and the HTO was collected in the bubblers. Tritium radioactivity in the bubblers was measured with a liquid scintillation counter.

The bubblers were replaced with new ones each time. It took about 2.5 hours per one recovery operation. The tritium radioactivity at the first recovery operation was 31.5 kBq in the bubbler. We also performed the second irradiation experiment to measure the tritium production value with the direct dissolution method. From the comparison between the tritium quantity recovered with the heating and the tritium production quantity measured with the direct dissolution, the tritium recovery ratio was deduced 0.98.

These results demonstrated that tritium produced in the sample pebble was completely emitted at 800°C and that helium gas contained hydrogen gas used as recovery gas could recover tritium without tritium accumulation in the pebble and pipe line. From these results, for the first time, we clarified that the lithium titanate pebble blanket had an extremely high feasibility of sufficient tritium recovery by heating to 800°C. As our next objectives, we are planning to perform recovery experiments under DT neutron irradiation.

FTP/3-3Ra

Plasma-Facing Materials Research for Fusion Reactors at FOM Rijnhuizen

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In next generation magnetic fusion devices such as ITER, plasma-facing materials are exposed to unprecedented high ion, power and neutron fluxes. Those extreme conditions cannot be recreated in current fusion devices from the tokamak type. The plasma-surface interaction is still an area of great uncertainty.

At FOM Rijnhuizen, linear plasma generators are used to investigate plasma-material interactions under high hydrogen ion flux-densities ($\leq 10^{25}$ m^{-2}s^{-1}) at low electron temperatures ($\leq$ 10 eV), similar to the conditions expected in the divertor of ITER. The incident ion fluxes result in power fluxes of $\geq$ 10 MW/m^2. A new linear plasma device, MAGNUM-PSI, is expected to begin regular plasma operations in the middle of 2010. This device can operate in steady-state with the use of a 3 T super-conducting magnet, and a plasma column diameter projected to 100 mm. In addition, experimental conditions can be varied over a wide range, such as different target materials, plasma temperatures, beam diameters, particle fluxes, inclination angles of target, background pressures, magnetic fields, etc., making MAGNUM-PSI an excellent test bed for high heat flux components of future fusion reactors.

Current research is performed on a smaller experiment, Pilot-PSI, which is limited to pulsed operation, a maximum magnetic field of 1.6 T and a narrow ($\approx$ 20 mm) column width. The research is primarily focused on carbon based materials and refractory metals. Erosion of materials, surface morphology changes as well as hydrogen implantation, diffusion and inventory in the materials are studied under fusion reactor conditions. The influence of neutron damages is studied by irradiation of the materials with high energy ions.

A research programme addressing those before mentioned issues is presented. An overview of the results obtained on Pilot-PSI with respect to carbon erosion, hydrogen retention in refractory metals and transient heat loads are presented.

FTP/3-3Rb

Effects of Plasma Interaction with Radiation Damaged Tungsten

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Tungsten is one of the important candidates to be the plasma facing material for ITER and that possible for future fusion reactors. It is important, therefore, to investigate experimentally effects of the fusion reactor operation on this material. The problem of damage generated in structure materials by fast neutrons will be a serious concern in DT fusion tokamak reactors. An experimental investigation of relation between the plasma erosion effect on PFMs and the level of radiation defects in material is being conducted on the facilities at Kurchatov Institute. Experiments are performed using the Kurchatov cyclotron providing high-energy particles to produce damage in material and the linear plasma machine LENTA simulating tokamak divertor plasma conditions. New results of the experiments with tungsten W (99, 95% wt) similar to the ITER candidate grade are presented in this paper. The work is focused on the material at high level of radiation damage. Emphasis of the plasma experiments is laid on evaluation of erosion under high plasma flux and deuterium trapping. The neutron effect modeling corresponding to fluence of 10^{26} neutron/m^2 becomes possible. Tungsten samples were irradiated on the cyclotron by He^{++} ions at the energy of $3.5 - 4$ MeV to produce damage in the material. Three high-energy ion fluences were reached in these irradiations: 10^{21}, 10^{22} and 3×10^{22} He^{++}/m^2. Then the damaged samples were subjected to multiple exposures to steady-state deuterium plasma on the LENTA device. The plasma ion energy on the surface was 250 eV to simulate the edge tokamak plasma. Total plasma fluence was $\sim 10^{26}$ m^{-2} of deuterium ions on the W surface. The evaluation of the erosion rate did not show relation to the presence of damages in the samples. On the other hand, significant surface microstructure modification after plasma exposure has been observed by comparison of the damaged and non-irradiated tungsten. A significant increase of the retained deuterium in the damaged tungsten structure (by an order of magnitude) has been detected by ERDA. Accumulation of helium was also detected in tungsten by Elastic Nuclear Backscattering. The developed approach appears efficient for the research of PFMs as to their service life and tritium hazardous effects in a fusion reactor.

FTP/3-4Ra

Heat-Resistant Ferritic-Martensitic Steel RUSFER-EK-181 (Fe-12Cr-2W-V-Ta-B) for Fusion Power Reactor

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The study of initial (unirradiated) functional properties of Russian RAFMS RUSFER-EK-181 (Fe-12Cr-2W-V-Ta-B) as advanced heat- and radiation-resistant structural material for fusion power reactors has been continued. RUSFER-EK-181 steel is related to precipitation hardening (nanostructured) materials type. The regularities of low temperature brittle fracture (crack growth resistance) of the steel at static and dynamic concentrated loads depending on the sizes of Charpy V-Noch (CVN) specimens, type of stress concentrators (V-notches or a fatigue crack) were investigated in the temperature range from −196°C to +100°C. Fracture toughness tests estimating KIC and JIC (static concentrated bending) were conducted. The ductile-to-brittle transition temperatures (DBTT) were determined in the range from −85°C to +35°C depending on the type of CVN-specimens and stress states (fatigue crack, central and side V-notches). The work of low temperature fracture of the steel depends on the type of the stress concentrators and specimen sizes and is governed by the elastic energy store and the conditions of plastic deformation in the near-surface layers of the specimens regulated by side notches. The marked level of permanent deformation and impact toughness (not less than $3 - 5$ J/cm^2) was observed at low temperatures (lower than DBTT). Short-term (yield strength, ultimate strength, elongation to rupture) and long-term (creep, diagram "stress vs time to rupture") properties, temperature conductivity, thermal conductivity and linear expansion of the steel were investigated in the temperature range to 750°C. Diffusion characteristics of self-point defects (vacancies and interstitial atoms) in iron crystal with dislocations of different types were calculated by the methods of multiscale modeling in the temperature range from room temperature to 1000K. Nuclear transmutation changes of the chemical composition and the structure and phase state (Schaeffler diagram) of the steel were calculated for the long-term irradiation in neutron spectrum of the fusion reactor DEMO-RF. Obtained data for functional properties of the RAFMS RUSFER-EK-181 characterize it as the nanostructured structural steel with a good heat resistant (up to 700°C), good resistance to low temperature embrittlement, sufficient radiation resistance of the chemical composition and the structure and phase state.

FTP/3-4Rb

Low Cycle Fatigue Properties of Reduced Activation Ferritic/Martensitic Steels after High Dose Neutron Irradiation

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The development and thorough characterization of DEMO relevant structural materials as well as their validation under fusion relevant conditions are the prerequisites for reliable design and safe and successful operation of the DEMO and belong to the key tasks within the European long term fusion R&D programme. The current work focuses on the Low Cycle Fatigue (LCF) behaviour of RAFM steels irradiated to a displacement damage doses up to 70 dpa at $330-337°$ C in the experimental fast reactor Bor 60 within the ARBOR-2 irradiation programme. The influence of the neutron irradiation on the fatigue behaviour was determined for the as-received EUROFER97 (980° C/0.5 h +760°C/1.5 h), pre-irradiation heat treated EUROFER97 HT (1040°C/0.5 h +760°C/1.5 h) and pre-irradiation heat treated F82H-mod (1040°C/38 min +750°C/2 h) steels. The strain controlled push-pull loading was performed with miniaturized cylindrical specimens at a constant temperature of 330°C with different total strain ranges (ε_{tot}) between 0.8 and 1.1% and at common strain rate of $3 \times 10^{-3}\ s^{-1}$. The comparison with the corresponding results in the reference unirradiated state was performed both for the adequate total and inelastic strain amplitudes. The neutron irradiation induced hardening can affect differently the fatigue behaviour of the irradiated specimens. The reduction of the inelastic strain in the irradiated state compared to the reference unirradiated state for common total strain amplitudes can yield increase of fatigue lifetime. The enhancement of the stress state for the adequate inelastic strain in contrast might accelerate the fatigue damage accumulation. Depending upon which of the two mentioned effects is dominant the neutron irradiation can either extend or reduce the fatigue lifetime compared to the reference unirradiated state. The experimental results on EUROFER97 and EUROFER97 HT support the above considerations. Most of the irradiated specimens show fatigue lifetimes which are comparable to the reference unirradiated state for adequate inelastic strains. In some cases, however, lifetime reduction is observable. F82H mod showed partly lifetime enhancement compared to the reference unirradiated state for adequate inelastic strains.

FTP/3-4Rc

Reduced Activation Ferritic/Martensitic Steel F82H for in-Vessel Components

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Recent results on the effects of irradiation on mechanical properties of F82H are described. The results revealed that optimization and tightening of the tempering condition improved the irradiation response with respect to toughness and ductility. The impact of irradiation effects on the structural integrity is evaluated and the feasibility of this steel for DEMO application is discussed. The reduced activation ferritic/martensitic (RAF/M) steel F82H is recognized to be a leading candidate material for in-vessel components in a future fusion DEMO reactor, as well as that for ITER test blanket modules (TBM) in Japan. Examination of the neutron irradiation response of RAF/M steels has been carried out mainly in JA-US collaborative experiments using the High Flux Isotope Reactor at Oak Ridge National Laboratory. F82H is the reference heat for the collaborative experiments. The results of recently completed post-irradiation examination demonstrated a remarkable improvement in the irradiation-induced degradation of fracture toughness after optimizing and tightening the heat treatment condition, as well as by addition of minor alloying element of tantalum. The degradation of fracture toughness was evaluated by the shift of ductile to brittle transition temperature (DBTT-shift). The optimization and tightening of tempering condition are also revealed to be effective for maintaining post-irradiation ductility. DBTT-shift and reduction of ductility by irradiation have been recognized to be critical issues for RAF/M steels; therefore, the beneficial effect of the optimization of tempering condition may be to extend the service conditions of RAF/M steels and seem to deliver more flexibility for DEMO design. In this paper, the current status of development of F82H is reported. Results of fracture toughness, tensile, creep and fatigue properties are introduced focusing on the improvement of the irradiation response of DBTT. The impact on structural integrity is also briefly discussed. Recent developments for the fabrication of components will be introduced. The methodology of improving post-irradiation toughness and ductility is also expected to be applicable for other RAF/M steels.

FTP/3-5Ra

Progress on Design and R&D of CN Solid Breeder TBM

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The current progress on design and R&D of Chinese helium-cooled solid breeder test blanket module HCSB TBM are presented. The updated design on structure, neutronics calculation, thermal-hydraulics and safety analysis have completed. In order to accommodate HCSB TBM system, the design and R&D of ancillary sub-systems have being developed. The R&D on the function materials, the structure material and the helium test loop construction are presented. The Chinese low-activation ferritic/martensitic steels CLF-1 as structural materials of HCSB TBM are developing towards industrially manufacture. The neutron multiplier Be and tritium breeder Li_4SiO_4 pebbles have been fabricated at laboratory level.

FTP/3-5Rb

Mock-up Fabrication and Component Tests for Water Cooled Ceramic Breeder Test Blanket Module

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The real-scale component mock-ups of the side wall and the pebble bed container have been successfully fabricated for the Test Blanket Module (TBM) with the water cooled ceramic breeder. The TBM has a box structure and contains the tritium breeder and neutron multiplier pebble beds inside. The bed containers with a box structure have cooling channels so that the breeder and multiplier materials are kept under the maximum allowable temperatures. The layers of the breeder and the neutron multiplier are alternately located in the radial direction. This structural concept leads to the effective tritium breeding and the isolation of the bred tritium from the beryllium layers.

For the side wall, the parallel multi-channel coolant paths are designed in order to reduce the pressure drop. The water flow test using the fabricated mock-up shows that the flow distribution is acceptable in view of the heat removal, even though the inner diameter of the manifold is limited within the thickness of the side wall. For the pebble bed container, the membrane structure is adopted to reduce the amount of the structural material leading to increase of the tritium breeding. Using the fibre laser, 1 mm thick tube and 1.5 mm thick plate are well welded to be the membrane structure. An adequate packing factor is obtained for the Li_2TiO_3 pebbles, and the pressure drop of He purge gas is confirmed to be reasonable. In addition, the side walls and the U-shaped first wall that was previously fabricated have been assembled into an open box with five faces. The heat

removal capability of the first wall mock-up was previously confirmed by the heat load test. Therefore, fabrication technologies and functions of the components in the TBM have been well validated. The achieved technologies can be applied to the fabrication and the assembly of the back wall, then the box structure of the TBM will promisingly be built in the next stage.

FTP/3-6Ra

NSTX Lithium Technologies and Their Impact on Boundary Control, Core Plasma Performance, and Operations

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Replenishable liquid lithium PFCs show promise towards resolution of density and impurity control, tritium and dust removal, and long-lifetime walls for diverted high power DT reactors by providing a low-Z, pumping, and self-healing plasma facing surface, and enabling a lithium wall fusion regime. Motivated by this potential, lithium pellet injection (LPI), evaporated lithium, and injected lithium powder have been used in succession to apply lithium coatings to graphite plasma facing components in NSTX high-power divertor plasma experiments. In 2005, following wall conditioning and LPI, discharges exhibited edge density reduction and performance improvements. Since 2006, first one, and now two lithium evaporators have been used routinely to evaporate lithium onto the lower divertor region at total rates of $10 - 70$ mg/min for periods $5 - 10$ min between discharges. Since 2008, prior to each discharge, the evaporators are withdrawn behind shutters. Significant improvements in the performance of NBI heated divertor discharges resulting from these lithium depositions were observed. In 2009, these evaporators were used for more than 65% of NSTX experiments. The improvements in NBI-heated divertor discharges resulting from these lithium depositions include: reduced edge recycling; increased energy confinement; suppression of ELMs; and decreases in the inductive flux consumption resulting in longer pulse lengths. The application of lithium has also reduced or eliminated the HeGDC previously required between discharges, thereby allowing for an increased duty cycle. Initial work with injecting fine lithium powder into the edge of NBI heated deuterium discharges using a recently developed piezoelectric resonant acoustic injector, yielded comparable changes in performance. Several technology issues encountered with lithium wall conditions are discussed. The next step in this work is to test a Liquid Lithium Divertor (LLD) surface that was installed recently on the outer part of the lower divertor. The LLD consists of a toroidal array of 20 cm wide plates with a 165 micron layer molybdenum with 45% porosity, plasma sprayed on a protective barrier of 0.25 mm stainless steel, bonded to a 2.2 cm thick copper substrate. The design, methods, and application of these lithium technologies will be presented.

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FTP/3-6Rb

Development and Experimental Study of Lithium based Plasma Facing Elements for Fusion Reactor Application

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Results of experimental study of lithium capillary-porous systems as the plasma facing material in tokomak conditions are considered. Impressive effect of lithium application in plasma performance improvement for domestic (T-11M, T-10)and foreign (FTU, NSTX, LTX, HT-7) fusion installations are observed. New results and plans on lithium experiments in FTU, KTM, T-11M tokamaks and stellarator TJ-II are presented.

FTP/P1 — Poster Session

FTP/P1-01

Thermal Diffusivity Measurement of Graphite Pebble Bed by Laser Flash Method

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In solid-type breeding blankets for fusion reactors, most of the concepts employ materials in the breeding zone to be packed as pebbles, which is called collectively as a pebble bed. For reliable blanket system design, thermo-mechanical properties of pebble beds have to be well understood and measured. Especially, in order to apply conventional continuum approach to predict thermo-mechanical behaviors of pebble beds, such as Finite Element Method (FEM), it is important to measure the effective properties of pebble beds as well as properties of the pebble itself. In the present work, thermal diffusivity of graphite pebble bed is measured. The graphite reflector is unique feature of Korean TBMs in that the amount of beryllium multiplier is considerably reduced by replacing some of it with the graphite reflector while tritium breeding capability remains almost the same. Therefore, establishing thermo-mechanical data for the graphite pebble bed is critical for Korean fusion programs. For measurements, Laser Flash Method (LFM), which is a non-contact and non-destructive method with covering a wide range of diffusivity with excellent accuracy, is applied and wall-influence effect on the minimum thickness required for the sample is investigated for future experiments.

FTP/P1-02

Surface Behavior in Deuterium Permeation through Erbium Oxide Coating

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Control of tritium permeation through structural materials is a key investigation for realization of an operative fuel cycle in fusion blanket concepts. Recent years, investigations on erbium oxide thin films have been demonstrated as a promising candidate for tritium permeation barrier. In this study, permeation measurements and structural analyses have been performed to clarify a surface behavior of deuterium permeation through the erbium oxide coatings deposited by a physical vapor deposition technique.

The erbium oxide coatings were deposited by filtered vacuum arc deposition on one side or both sides of reduced activation ferritic/martensitic F82H and JLF-1 steel substrates at room temperature. The permeability of the samples with a 2.6 micron-thick erbium oxide layer on one side and 1.3 micron-thick layers on both sides has been measured by a gas-driven deuterium permeation apparatus. Permeation reduction factors of 1000 and 10 000 have been evaluated respectively, which means the both-sides coated sample has shown one order of magnitude lower permeation flux than that coated on one side in spite of the same effective thickness. In addition, the activation energies of permeation through each sample are estimated 81 and 117 kJ/mol by temperature dependence of the permeability. That indicates the surface reactions on the coating possess higher activation energies than diffusion process.

The deuterium permeation behavior has been examined with 1.3 micron-thick coatings deposited on one side facing to the high and low deuterium pressure sides. The coating on the high deuterium pressure side has 7 times higher permeation reduction factor than that of the coating facing to the low pressure side. Additionally, X-ray diffraction spectra of the coated samples after the permeation measurements have shown an orientation of the erbium oxide on the coating facing to the high deuterium pressure side, which can cause the difference of permeability. Furthermore, compared to the result of the both-sides coated sample, the value of permeation reduction factor can be described not by addition but by multiplication of those of the coatings facing to each side. This result suggests an important finding that the surface reaction probability affects dominantly the deuterium permeation through the both-coated sample.

FTP/P1-03

Manufacturing and Heat Transfer Tests of a Rectangular Tray-Type Tritium Getter Bed

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To verify the feasibility of the design of a rectangular tray-type tritium getter bed, a manufacturing of a mockup and heat transfer test were performed at KAERI. The storage bed was fabricated by brazing the tray plate, heater tube and helium loop. Brazing enables improved heat transfer for heating and cooling the plates. Feed-through connections of the heater line and thermocouple line of the primary vessel enable the prevention of tritium leakage into the secondary vessel. The circumferential thermal reflectors and circular plate-type thermal reflectors for the primary vessel were fabricated to reduce heat transfer from the primary vessel to the secondary vessel, and thermal loss. Vacuum and leak inspections on the primary and secondary vessels for the fabricated rectangular tray-type bed were performed. These results were proved to have no leaks on the tubes of the primary and secondary vessels. Furthermore, the bed showed rapid heating-up characteristics, which is desirable for a rapid delivery of hydrogen isotopes by desorption from the metal hydride.

FTP/P1-04

Considerations towards the Fuel Cycle of a Steady-State DT Fusion Device

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Control and management of the fuel and fusion product streams is one of the most difficult issues for fusion energy development. The closed fusion fuel cycle comprises an inner and an outer part. The inner part includes the fuelling systems (gas puffing, pellet injection, etc.), the vacuum pumping systems and the tritium plant systems (fuel cleanup, isotope separation, storage and delivery). The outer part includes the breeding blankets and their tritium extraction and recovery systems. The challenges of these components are given by the need to handle large flows of tritium resulting in strong requirements on containment and accountancy. ITER will in some areas serve as a good basis for scale-up to a fusion reactor, in many other areas not. In all latter cases, new technologies have to be developed. This paper will go through the current state of technology and identify

the areas which are considered to require essential supporting R&D towards a functional and technically ready system for a reactor. Furthermore, recent experimental results are presented and discussed and R&D routes are identified. This paper will also recommend topical areas which need focussed efforts, such as D-T-mix and burn-up fraction increase and illustrate the needs to provide better and well validated modelling tools. Efforts under EFDA are currently on the way to move in this direction.

FTP/P1-05

Development of a Neutral Particle Flow Fueling System by using a Compact Torus Plasma Injector for LHD

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The Compact toroid (CT) fueler of SPICA (SPheromak Injector using Conical Accelerator) for LHD has been developed at NIFS. Recently, in order to apply CT injection technique to more effective fueling, production and injection of super-high speed neutral particle flow (NPF) have been studied. The NPF injection has a performance advantage over supersonic gas jet and can play a specific role beyond neutral beam injection (NBI). We have then proposed a fueling system of CT-based NPF injection. The system may allow us to make various plasma controls. The SPICA injector had two-stage coaxial electrodes for CT formation and acceleration. However, for practical setup of SPICA on LHD, the system should be simple and reliable for easy operation and maintenance. We thus attempted single-stage operation by connecting only the acceleration bank unit to both electrodes. By using the simple SPICA injector, we launched the study on production of super-high speed NPF in the following scenario; SPICA accelerates a CT plasmoid and injects it into a long drift tube as a neutralizer cell filled with hydrogen gas, then super-high speed NPF is produced through charge-exchange (CX) reaction between CT plasma and neutral gas. In the first phase of the experiment, we have investigated the performance of the simple SPICA. CT speed was obtained to be ~ 100 km/s and CT density was up to 1×10^{22} m^{-3}. Although the speed is rather low, the density is remarkably high. If the CT plasmoid is completely neutralized, the particle inventory of NPF is estimated to be 2×10^{20} from the FWHM of an electron density signal. This can provide a density increment of 7×10^{18} m^{-3} in an LHD plasma. The penetration depth of NPF is estimated roughly at 6 mm, which is several hundredths of that of NBI. The input power is, however, calculated to be 32 MW since the NPF is injected at a short pulse of about 100×10^{-6} s. The injector corresponds to a NBI at 100 eV and 320 kA. It would be almost impossible to develop such a NBI. In addition to the experiment, numerical calculation has been made to understand the neutralization process and investigate the conditions for high neutralization efficiency. In the follow-on work, we intend to make quantitative

measurement of neutralization efficiency, and also compare the experimental result with the calculation.

FTP/P1-06

Gas Chromatography Separation of H_2-D_2-Ar using Pd/K

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A new type of gas chromatographic separation of hydrogen isotopes has been developed based on hydrogen isotope effect of palladium, which could separate H_2-D_2-Ar mixtures and HD into H_2, D_2 and Ar near room temperature with a simply operation procedure without introducing any other replacement gases. This method was investigated using 63 g Pd/K as the functional material to separate H_2-D_2-Ar with two cascaded columns. The palladium content of the Pd/K was about 38.1 ± 0.1 wt% with the particle sizes of $\sim 200 - 350\,\mu$m. The H-D separation effect was studied under different operation conditions including the temperatures of the columns, the flow rates of the feed gases and the contents of hydrogen isotopes. The results show that the separation of protium-deuterium will be better when the protium and deuterium contents in raw gases are within 6 wt%, with a flow rate of 10 Ncm^3/min at lower temperatures. The new chromatographic separation technique has a potential for separation and recovery of hydrogen isotopes.

FTP/P1-07

Variation of PCT Isotherm in the Disproportionated ZrCo

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The PCT isotherm of the ZrCo hydride system having a partial portion of disproportionation phase was studied to measure the complex phase equilibrium and to compare the difference of the partially deactivated metal hydride system with the normal ZrCo hydride component system in the specific temperature range above 350°C. A partial portion of the disproportionation in the ZrCo hydride was estimated by Sivert PCT apparatus, XRD and EDX measurements and so forth. As a result, a modification of H_2/ZrCo atomic ratio in the PCT diagram was adopted to depict a new PCT isotherm for the partially disproportionated metal hydride system, while the effect of disproportionation phase on the PCT isotherm of the ZrCo hydride system was directly related to the disproportionated amount of ZrH_2 and $ZrCo_2$.

FTP/P1-08

A-C:H Film Removal from and Oxidation of W and Mo in H_2/Air Glow and Afterglow Discharge

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Hydrocarbon and oxide films covering plasma-facing materials can be a source of species penetrating into plasma. Therefore, the material surfaces must be as clean as possible. The aim of the work is to develop an effective oxygen-containing gas discharge technology for removal of hydrocarbon deposits from metallic substrates and minimal oxidation of their surfaces. In the experiments a stream technique has been used in which a cylindrical quartz tube was a main part. A direct current glow discharge was as a source of different species. Hard and soft a-C:H films have been deposited on W and Mo. Before and after the experiments, C, N, and O areal densities in the samples were measured by electron probe microanalysis (EPMA). The film erosion experiments in the H_2/air (H_2 - 7 sccm, air - 1.38 sccm) discharge (320K) have shown that erosion rates increased about 6 times as compared to pure H_2 discharge and were about 43, 6 and 1.5 nm/min in the cathode, positive column and afterglow regions, respectively. After removal of the a-C:H films the carbon traces on the W and Mo substrates were not observed by EPMA. At the same time in the cathode area the nitrogen and oxygen areal densities in the surface layers were about 0.002 and 0.0045 mg/cm^2, respectively. In the positive column area the nitrogen and oxygen areal densities were about 0.002 mg/cm^2. It has been shown that after removal of the a-C:H films a primary ($\sim$ 1 nm thick) continuous amorphous oxide film transformed into secondary island-like oxide films. The polycrystalline island-like oxide films were generated due to recrystallization of the primary oxide film. Metal oxidation suppression was caused by reactions of oxygen ion neutralization and atomic oxygen recombination on metals. In products of a-C:H film erosion CO, CO_2, H_2O, NH_3, and HCN have been found by a quadrupole mass-spectrometer. Air presence in low-temperature hydrogen plasma results in formation of stationary thin oxide films that can be reduced during about one day exposure in hydrogen discharge. The technique suggested is one of possible variants of an effective gas-discharge technology for a-C:H film removal from metal surfaces.

FTP/P1-09

Measurement of Deuterium Diffusion and Permeation in Several Stainless Steels

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It is necessary for safety and circumstance influence evaluation to make sure that the hydrogen isotopes permeate through stainless steel used in tritium system. A dynamic quadrupole mass spectroscopy method was adopted to measure deuterium permeation and diffusion through 1Cr18Ni9Ti, HR1, HR2 stainless steels. It was indicated that this method had many advantages over other methods, such as high sensitiveness (up to 10^{-9} Pa·l·s^{-1}), fast response, and high precision with a combined standard uncertainty of 18%. The permeability and diffusivity of deuterium in these stainless steels were measured over the temperature range of $\sim 473-823$K. The influences, such as temperature, pressure, oxidation of surface, welding and isotopes were investigated. The permeation activation energy of deuterium for 1Cr18Ni9Ti, HR1, HR2 stainless steels are 50.69 kJ, 55.66 kJ, 50.81 kJ, respectively. The diffusion activation energy of deuterium for 1Cr18Ni9Ti, HR1, HR2 stainless steels are 50.69 kJ, 55.66 kJ, 50.81 kJ, respectively. The flux of permeation is in proportion to the square root of pressure. The permeability of welding zone is $1-2$ times higher than of base steel. The flux of permeation decreases by $4-10$ times for surface oxidation. However, the decreasing effect disappears when the samples were heat to 873K.

FTP/P1-10

A Fusion Neutron Source for the Incineration of Radioactive Waste based on the Gas Dynamic Trap

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The topic transmutation of long-lived radioactive nuclear waste, including plutonium, minor actinides and fission products, represents a highly important problem of nuclear technology and is presently studied worldwide in large-scale. Sub-critical systems seem to be a promising option for efficiently burning plutonium and minor actinides provided a sufficiently high-intense neutron source is available.

For a number of years the Budker Institute of Nuclear Physics (Russia) in collaboration with the Russian and European organizations developed the project of a 14 MeV neutron source for fusion material irradiation and other applications. The projected plasma type neutron source is based on the Gas Dynamic Trap (GDT) which is a special magnetic mirror system for the plasma confinement. The current work further elaborates the concept of a GDT-based neutron source for nuclear applications. The paper presents a 3D numerical model of the neutron source for the transmutation of long-lived radioactive waste

in spent nuclear fuel. The plasma physics calculations of the neutron source's parameters have been performed with the Monte-Carlo method using the Integrated Transport Code System (ITCS). ITCS was developed for GDT simulations and includes different modules for plasma, particles transport and neutron production modeling. New physical phenomena such as a vortex confinement, improved axial confinement, ambipolar plugging, high beta etc. were included in these simulations. The experimental and theoretical foundations of these phenomena were obtained in the GDT-U experimental facility in the Budker Institute.

As a result, a new improved version of the GDT type fusion neutron source is proposed and numerically simulated. The proposed neutron source has two n-zones of 2 m length with a neutron power of 1.6 MW/m and a neutron production rate up to 1.5×10^{18} n/s each. This source can be used for application to the Fusion Driven System (FDS) for the burning of MA in spent nuclear fuel. A system with one fusion GDT driver for two sub-critical burners placed around the neutron emission zones is analyzed in this paper. Results of the nuclear analyses based on neutron transport calculations with the MCNP Monte Carlo code will be presented.

FTP/P1-11

Nearer Term Fission-Fusion Hybrids: Recent Results

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Two pre-conceptual designs of fission-fusion hybrid reactors, with multiple novel elements to enable nearer term realization, are described. The centerpiece of the first design (Compact Hybrid, University of Texas) is an ST based, replaceable, modular high power density Compact Fusion Neutron Source (CFNS), enabled by the Super-X divertor geometry. The second design (SABR, Georgia Tech) invokes a standard tokamak, based on ITER physics and technology. Both designs choose dimensionless plasma parameters (beta normal and H) in the range anticipated in ITER, and both are designed to be self-sufficient in tritium.

The SABR design combines a sodium-cooled fast fission blanket with the tokamak. A fission power of 3000 MWth provides a significant transmutation rate, and existing balance of plant equipment can be used. The fuel assembly (Zr40-TRU metal arranged to form an annular reactor core surrounding the toroidal plasma in 4 rings) is shuffled. One SABR, operating at 76% availability, would fully support three 1000 MWe LWRs. Most plasma and technology parameters for the neutron source are within the range which will be demonstrated in ITER, so SABR could be deployed by about 2040.

Mechanical and electromagnetic independence of the fusion and fission systems, a hallmark of the Compact Hybrid, is achieved by placing the fission blanket outside the TF coils. Modularity and replaceability of the CFNS will let independent, parallel developments of the fusion and fission components. Divertor operation, He exhaust, current drive efficiency, magnet thermo-mechanics, and Li operation are analyzed with state of the art codes.

A 100 MW CFNS facilitates incineration of problematic isotopes allowing high support ratio (S) fuel cycles. Using either fast spectrum or thermal spectrum fission blankets, a single Compact Hybrid could support as many as twenty or more fission reactors both in waste incineration or fuel production.

The compact, low mass CFNS enables using it as a replaceable module, giving high plant availability even with an individual CFNS availability $< 50\%$. Components exposed to 14 MeV neutrons might be replaced as often as every scheduled plant outage, greatly reducing neutron damage. This substantially reduces technological requirements for a CFNS (in availability and neutron damage), enabling realization of a hybrid with less time, cost and risk.

FTP/P1-12

Fissile Fuel Breeding and Actinide Transmutation in an Inertial Fusion Energy Reactor

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Macro scale fusion projects are ITER on magnetic fusion energy (MFE) and NIF on inertial fusion energy (IFE). While ITER is still on design stage, NIF will become fully operational early 2011.

Preliminary design studies on a Laser Inertial Confinement Fusion Fission Energy (LIFE) reactor concept have been initiated at LLNL, based on NIF. LIFE has a spherical fusion chamber of 5 m diameter, a thin LiPb coolant behind an ODS first wall, followed by a beryllium multiplier, fission blanket and graphite reflector. However, beryllium multiplier will soften neutron spectrum leading to local heat peaks.

In the present work, a modified LIFE blanket has been investigated consisting of ODS-first wall (2 cm); Li17Pb83 as neutron multiplier+coolant; ODS-separator (2 cm); Flibe (Li_2BeF_4) coolant and fission zone; ODS-separator (2 cm); graphite reflector. The study covers fissile fuel breeding and minor actinide transmutation. Tritium breeding ratio (TBR) is kept > 1.05. Thicknesses of blanket zones have been determined as 10 cm Li17Pb83; 50 cm Flibe and 30 cm graphite reflector, leading to $TBR = 1.134$ without fissionable material. Addition of fissionable fuel decreases TBR. A volume fraction of 2% TRISO particles with nat-UC in molten salt yields $TBR = 1.0514$.

Minor actinide have superior nuclear properties at high energies and lead to higher neutron multiplication under 14 MeV fusion neutron irradiation. One obtains $TBR = 1.05$ by

11 vol.% of TRISO particles containing minor actinides. Furthermore, time evolution of fissile and fusile fuel breeding, figure-of-merit (plutonium production/MW-fusion power), actinide transmutation over operation, material damage on structures with respect to DPA, He and H production, and Shallow burial index for final decommissioning are being evaluated.

FTP/P1-13

A Fusion-Fission Reactor Concept based on Viable Technologies

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The world needs a great deal of carbon free energy for civilization to continue. Nuclear power is attractive for helping cut carbon emissions and reducing imports of fossil fuel. It is commonly realized that it needs hard work before pure fusion energy could be commercially and economically utilized. Some countries are speeding up the development of their fission industry. In China, the government has decided to develop nuclear power with a mid-term target of ~40 GWe in 2020. If only PWR is used to meet the huge nuclear capacity requirement, there may be a shortage of fissile uranium and an increase of long-lived nuclear wastes. Therefore, any activity to solve the problems has been welcome. A lot of research activities had been done to evaluate the possibility of the hybrid systems in the world, however, most of them were based on advanced fusion and fission technologies.

In this contribution, three types of fusion-fission hybrid reactor concepts, i.e. the energy multiplier named FDS-EM, the fuel breeder named FDS-FB, waste transmuter named FDS-WT, have been proposed for the re-examination of feasibility, capability and safety and environmental potential of fission-fusion hybrid systems. Then based on the re-evaluation activity, a multi-functional fusion-fission reactor concept named FDS-MF simultaneously for nuclear waste transmutation, fissile fuel breeding and thermal energy production based on viable technologies i.e. available or limitedly extrapolated nuclear, processing and fusion technologies is proposed. The tokamak can be designed based on relatively easy-achieved plasma parameters extrapolated from the successful operation of the Experimental Advanced Superconducting Tokamak (EAST) in China and other tokamaks in the world, and the subcritical blanket can be designed based on the well-developed technology of PWR. The design and optimization of fusion plasma core parameters, fission blanket and fuel cycle have been presented.

And the preliminary performance analyses covering neutronics, thermal-hydraulics, thermomechanics, and electricity economic assessment have been carried out. The calculation results show that the concept based on viable technologies can meet the requirements of tritium self-sufficiency, sufficient energy gain and economy attractiveness of produced electricity, and operate for at least 10 years without fuel unloading and reloading.

FTP/P1-14

Study on Fission Blanket Fuel Cycling of a Fusion-Fission Hybrid Energy Generation System

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Direct application of ITER-scale tokamak as a neutron driver in a subcritical fusion-fission hybrid reactor to generate electric power is of great potential in predictable future. This paper reports a primary study on neutronic and fuel cycle behaviors of a fission blanket for a new type of fusion-driven system (FDS), namely a subcritical fusion-fission hybrid reactor for electric power generation aiming at energy generation fueled with natural or depleted uranium. Using COUPLE2 developed at INET of Tsinghua University by coupling the MCNP code with the ORIGEN code to study the neutronic behavior and the refueling scheme, this paper focuses on refueling scheme of the fissionable fuel while keeping some important parameters such as tritium breeding ratio (TBR) and energy gain. Different fission fuels, coolants and their volumetric ratios arranged in the fission blanket satisfy the requirements for power generation. The results show that soft neutron spectrum with optimized fuel to moderator ratio can yield an energy amplifying factor of $M > 20$ while maintaining the TBR > 1.1 and the CR > 1 (the conversion ratio of fissile materials) in a reasonably long refueling cycle. Using an in-site fuel recycle plant, it will be an attractive way to realize the goal of burning ^{238}U with such a new type of fusion-fission hybrid reactor system to generate electric power.

FTP/P1-15

Micro-Mechanical Testing and Nanoindentation of Tungsten Alloys for Fusion Applications

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There is a great deal of interest in using tungsten alloys as critical plasma facing components in both the divertor and blanket in future nuclear power plants. For this to occur the mechanical properties, and how they change under the extreme conditions faced in the power plant must be understood. Of particular importance are the brittle to ductile transition (BDT) and the effect of neutron damage on mechanical properties. This paper will discuss novel experimental work carried out at the University of Oxford to measure these. Recently much work has been performed in Oxford using small scale 4 point mechanical tests to measure the brittle to ductile transition temperature (BDTT) in a range of pure tungsten materials. Tests are now being performed on W 1 wt%Ta and W 5 wt%Ta alloys at temperatures from room temperature to 900°C. This allows the critical DBTT to be measured. Results from this testing programme will be reported at the meeting.

Neutron irradiation in a fusion reactor will cause a large amount of damage to the microstructure and also transmutation of W to Re and Os, producing He, and hence degrade the mechanical properties of the material. Specimens of pure tungsten, W-5wt%Ta, W-1wt%Ta and W-5wt%Re have been irradiated using an ion-beam facility at the University of Surrey, UK. 2 MeV tungsten ions were used, at 500°C, producing a damaged layer of depth $\sim$ 200 nm with doses aimed at generating damage levels of 5 dpa and 15 dpa. These damaged layers have been tested using nanoindentation which shows an increase in hardness in the implanted layer. However due to ill defined stress state around the indenter properties such as yield stress cannot be measured. By using focused ion beam machining micro-cantilevers can be machined in the damaged layer and bent using a nanoindenter. This will allow yield stress and work hardening of the ion implanted damaged layer to be measured directly without influence of the underlying material. Differences in the mechanical properties of implanted W-Ta and W-Re alloys are also being investigated.

FTP/P1-16

From Materials Development to their Test in IFMIF: an Overview

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R&D activities on fusion reactor materials in Switzerland focus on (1) the development and characterization of advanced metallic materials for structural applications in plasma facing (first wall, divertor) and breeding blanket components of the fusion power reactors, in particular oxide dispersion strengthened (ODS) reduced activation ferritic (RAF) steels and tungsten-base materials, (2) the modelling of radiation damage and radiation effects, and (3) small specimen test technology for the future International Fusion Materials Irradiation Facility (IFMIF). The scientific approach being used is based on investigating the structure/mechanics relationships at different length scales (nano-, micro-, meso-, and macroscopic), before and after irradiation, by using a wide range of experimental and numerical tools. For simulating experimentally the effects of 14 MeV neutrons, irradiations are being performed with a mixed spectrum of high-energy protons and spallation neutrons in the Swiss Spallation Neutron Source (SINQ) located at the Paul Scherrer Institute (PSI). The main objectives, examples of recent results and future activities are described in the case of the three R&D areas mentioned just above.

FTP/P1-17

Radiation Damage from Atomic to Meso-Scales in Extreme Environments

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A foreboding materials challenge is to be able to withstand the $10 - 15$ MW-year/m^2 neutron and heat fluence expected in the first wall and blanket structural materials of a fusion reactor. Overcoming radiation damage degradation is a key rate-controlling step in fusion materials development. New science, approaches, and facilities are needed at multiple scales. One of several new Energy Frontier Research Centers is one Los Alamos leads for "Materials at Irradiation and Mechanical Extremes". Its objective is to understand, at the atomic scale, the behavior of materials subject to extreme radiation doses and mechanical stress in order to synthesize new materials that can tolerate such conditions. Current computational and experimental research into new materials will be described.

The Matter Radiation Interactions in Extremes (MaRIE) concept is a National User Facility to realize the vision of 21st century materials research and development. A key extreme environment for future energy security is intense radiation. The Fission and Fusion Materials Facility (F3) segment of MaRIE proposes to use the present proton linac at Los Alamos with a power upgrade to drive a spallation neutron source that can provide the required radiation environment. Importantly, F3 would also provide the capability for in-situ measurements of transient radiation damage, using unique x-ray and charged particle radiography diagnostics. Coupled with integrated synthesis and characterization capability, these would reveal not only the atomic scale origins of radiation effects but also their meso-scale consequences. The mission of the facility would be to study material science in radiation extreme environments to provide predictive capability of material performance to allow certification of new nuclear systems. It is not intended for large-volume component testing. The radiation environment possible will be described.

FTP/P1-18

Development of Low Activation Superconducting Material for the Feedback Coil operated around Core D-T Plasma

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The feedback coil is one of the important components to improve plasma confinement. The feedback coil must be installed near the core plasma and it will be affected by the heavy D-T neutron irradiation. We proposed MgB_2 superconducting wire as the candidate material for the feedback coil around core D-T plasma. MgB_2 superconductor is one of the "Low activation superconductors" because the half-life of the induced radio-activity

on the MgB_2 superconductors is much shorter than that of Nb-based superconductors. We succeeded to fabricate 100 m class long and higher critical current density (Jc) MgB_2 mono-cored wire for fusion applications. Jc property of long MgB_2 wire was increased remarkably and its coil had 2.62 T of central magnetic field. We found that MgB_2 wire became alternative materials to the commercial Nb-Ti wire under the neutron irradiation environment and it has the large potential as "Low activation superconductor" to applying for advanced fusion reactor. In the case of the feedback coil around D-T plasma, the nuclear heat generation will be caused by the D-T neutron irradiation. The heat load of the correction coil is increased by the nuclear heat generation and the coolant temperature will be elevated by the heat load. The temperature margin between Tc value and operation temperature is one of the important factors to select superconducting materials of the feedback coil. The Ic-B property of MgB_2 wire was confirmed at 20 K of higher coolant temperature. Considering of the heat load by the nuclear heat generation in MgB_2 compound, the nuclear heat generation of MgB_2 compound was decreased remarkably from 2.58 to 0.127 mW/cm^3 if the isotope adjustment from ^{10}B to ^{11}B was carried out. This will be able to contribute the large tolerance of the operation temperature and coil design compared with Nb-Ti and Nb_3Sn wire.

FTP/P1-19

Engineering Design and Construction of IFMIF/EVEDA Lithium Test Loop

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Engineering design of IFMIF/EVEDA Lithium (Li) Test Loop, which is a proto-type of a Li target facility of International Fusion Materials Irradiation Facility (IFMIF), was completed and its construction was started under Engineering Validation and Engineering Design Activity (EVEDA) for IFMIF as one of ITER "Broader Approach" (BA) activities. Key issues to be validated in this Li loop are feasibility of hydraulic stability of a liquid metal Li flow as a deuteron beam target (Li target), and feasibility of Li purification systems. IFMIF/EVEDA Li Test Loop was designed to consist of two major Li circulation loops, which are a main and a purification loop including an impurity monitoring loop. The main loop was designed to circulate Li and form the Li target in equipment called target assembly which consists of a nozzle and a back plate. In this design, the target is a thin high-speed free-surface flow with 25 mm depth and 100 mm width produced by the nozzle, and flows along the back plate at the velocity up to 20 m/s (maximum design velocity). As for target assembly, two types of material and back plate were selected and designed: SS316L with an integrated back plate and F82H with a replaceable back

plate. The purification loop was designed to simulate the purification system envisaged in IFMIF and to install a cold trap; a nitrogen hot trap; a hydrogen hot trap. An existing building in a JAEA O-arai site was selected as the construction site for the EVEDA Li Test Loop. As of March 2010, the mount of the loop was almost completed, and ready to install equipments fabricated in factories.

FTP/P1-20

Flow Assisted Corrosion and Erosion-Corrosion of RAFM Steel in Liquid Breeders

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Study on flow assisted corrosion (FAC) and erosion-corrosion of RAFM JLF-1 steel (Fe-9Cr-2W-0.1C) in liquid breeders of Li, Pb-17Li and Flinak was carried out. It was found that the alloying element of Fe and Cr in the JLF-1 steel was commonly dissolved into these melts. The compatibility model of the JLF-1 steel in liquid breeders was developed. The mass loss of the specimens in the corrosion experiments was evaluated by the model. The effect of erosion-corrosion on the total mass loss of the steel in the liquid metals could be larger than that of FAC estimated by mass transfer calculation. The mass loss of the steel by electrochemical corrosion might be larger than that by the FAC in the Flinak.

FTP/P1-21

Corrosion Control of Er_2O_3 in Li as Insulating Material for Liquid Li Blanket System

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Blanket is one of the key components of fusion reactor to achieve sustainable tritium supply and economic rationality. Among blanket concepts, the liquid Li blanket system is regarded as an advanced blanket concept for its advantages such as high tritium breeding ratio without neutron multiplier, excellent heat transfer characteristics of Li and low tritium leakage without permeation barrier. These characteristics lead to a simple and economical blanket design. The crucial issue for this blanket concept is the MHD pressure drop. Fabrication of insulating coating which electrically decouples the Li from the structural metal is considered a solution for this problem. Among candidate materials for MHD coating, Er_2O_3 is a promising coating material due to its relatively good compatibility with Li and high electrical resistivity. However, detailed corrosion behavior of Er_2O_3 in Li is not known and it causes a difficulty to design Li blanket system.

Author's previously study showed Er_2O_3 forms $LErO_2$ in high temperature Li. In this paper we evaluate the effect of corrosion to the insulating property of Er_2O_3 from electrical resistivity of $LiErO_2$. The $LiErO_2$ was synthesized from Er_2O_3 and Li_2O in Li. The electrical resistivity of $LiErO_2$ is lower than that of Er_2O_3 and corrosion causes degradation of insulating property of the MHD coating.

Next, O kinetics in Li-Er system is investigated by immersing Er and Er_2O_3 into Li. O concentration in Li and test temperature are changed to investigate the chemical equilibrium and reaction speed. O was supplied to Li from gas phase. The Er_2O_3 was stable in certain O concentration range, and it became $LiErO_2$ if the concentration is higher than the upper limitation.

The corrosion of Er_2O_3 is severe if the O concentration is higher than limitation and the concentration must be controlled in Li blanket system. The appropriate O controlling method for Li blanket system is discussed in the paper.

FTP/P1-22

14 MeV Neutron Irradiation Effect on Superconducting Properties of Nb_3Sn Strand for Fusion Magnet

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Large scale plasma devices such as ITER and JT-60SA have been designed and the constructions have started. Also, design activity of helical reactor is carried out and the first plasma is anticipated to be realized in 2030s. The fusion reactor will generate a lot of neutrons and the superconducting magnets will be irradiated with streaming neutrons out of the ports and penetrating ones through the blanket and the vacuum vessel. The fast neutron changes the superconducting property by the knock-on effect.

This paper deals with the irradiation effect of 14 MeV neutron on the superconductivity of Nb_3Sn strand. The published papers show the change in the critical current by irradiation at a certain fixed magnetic field. But this study shows the dependence curve of the critical current on the magnetic field up to the critical magnetic field after irradiation. This is the world first result and the mechanism for the increase of the critical current is discussed.

FTP/P1-23

Status of Japanese Design and Validation Activities of Test Facilities in IFMIF/ EVEDA

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Under Broader Approach (BA) Agreement between EURATOM and Japan, IFMIF/ EVEDA (International Fusion Materials Irradiation Facility/Engineering Validation and Engineering Design Activities) has been performing from a middle of 2007. The IFMIF has three main facilities such as the accelerator Facility, Li Target Facility and Test Facilities. A previous design report of IFMIF was summarized in IFMIF comprehensive design report. The present EVEDA phase aims at producing a detailed, complete and fully integrated engineering design of IFMIF. For the first two and half years, the conceptual design was performed, and preliminary designs of engineering design have been prepared or are under preparation. Recent progress in Japanese engineering design and validation activities of test facilities in IFMIF/EVEDA project is summarized in this paper.

In the design of post irradiation examination (PIE) Facilities, functional analyses, evaluation of work process, and materials handling, including the specimen reloading in the capsule of HFTM which is required to obtain high doses up to about 150 dpa, for the materials transfer from test cell area such as HFTM and back plate of Li target to PIE facilities is very important. Therefore, these functional analyses for the design of PIE facility and schematic layout design of PIE facility were performed. In the evaluation of Japanese High Flux Test Module (HFTM) with the heater-integrated capsules to enable irradiation temperatures up to 1000°C, the manufacture technologies and performance tests for heater system of the irradiation capsule of HFTM was examined. For the materials of the heater, two materials, i.e., W-3%Re alloy and/or SiC/SiCf composite, were selected by the main reasons of high temperature materials, fabrication technology and some suitable properties such as resistance of thermal shock, high temperature re-crystallization, ductility, resistance of irradiation degradation, and low-activation materials. In small specimen test technique (SSTT), some subjects was examined for the methodology enabling the extrapolation from the small specimen data to standard specimen data, and some experimental data were obtained for standardization of test methodology and specimens' size and shape for fatigue, fracture toughness and crack growth in a F82H steel.

FTP/P1-24

Research and Development of Reduced Activation Ferritic/Martensitic Steel CLF-1 in SWIP

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Because of the good industrial bases and the superior resistance for irradiation, reduced activation ferritic/martensitic (RAFM) steel is recognized as the primary structural material for ITER test blanket modules (TBM) and a DEMOnstration reactor. In China, one of the basic options of the blanket module concept to be tested in ITER is helium cooled solid breeder (HCSB) with the RAFM steel as the reference structural material. To provide material and property database for the design and fabrication of the ITER HCSB TBM, a new type of RAFM steel CLF-1 was developed and characterized by South Western Institute of Physics in China. In this paper, recent progress in SWIP research on RAFM steel CLF-1 R&D is reviewed with a focus on ITER-TBM design and fabrication.

A new heat of 350 kg of CLF-1 steel was produced recently and different product forms (plates, rods and welding wires) were manufactured. Recent advances in the fields of steel development, mainly on the melting and processing techniques, composition optimization and thermo-mechanical treatment were addressed. The properties database and technical information required for blanket design and fabrication were derived.

From the tensile and creep properties test, the design allowable stresses are derived. From the Charpy impact test, ductile to brittle transition temperature (DBTT) are evaluated. From the physical properties databases, density, modulus of elasticity, thermal conductivity, thermal diffusivity, specific heat, linear expansion coefficients are derived. The effect of thermal ageing on the microstructure and properties was investigated to study the stability under high temperature for long periods of time. In addition, the efforts to characterize the weldability of CLF-1 using tungsten-inert-gas (TIG) method for the fabrication of TBM were also introduced.

FTP/P1-25

Why using Laser for Dust Removal from Tokamaks

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During ITER lifetime, the interaction of the all types of plasmas with the Plasma Facing Components (PFCs) will lead to the generation of dusts from various sizes (10 nm to 100 microns), shapes and compositions (Be, C, W and metallic impurities). These dusts will be activated, tritiated and potentially chemically toxic. To keep the amount of dust below these safety limits, a cleaning procedure has to be defined. In this study we investigate the capability of the laser cleaning process to be applied to this specific

application. Two laser-induced mobilization techniques have been studied. First, a direct irradiation of the substrate usually named Dry Laser Cleaning (DLC) and then Laser-induced Shockwave Cleaning (LSC) technique based on the generation of shockwave to blow the particles. In DLC system, specific studies undertaken on ITER-like particles provided clear view of ejection mechanisms and have shown the efficiency of such cleaning process. For carbon dust, we demonstrated that the main part of the ablation products is composed of atoms, whereas the metallic particles are ejected in a single part without any modification of their morphological properties. In this latter case, the electrostatic forces generated by the photoelectrons extracted from the particles are suggested as the main mechanism of the dust ejection. In LSC process, the focalisation of laser beam in gas or on a surface induces the gas breakdown and then the generation of a shockwave which blows the dusts. This technique appears very efficient to remove dust from castellations. In conclusion, the results obtained in the frame of this study provide a clear view of the ability of laser process to remove ITER-like dust from hot surfaces. They also permit to define the optimum operating conditions and the limitations of such technique.

FTP/P1-26

Recent Results on ICRF Assisted Wall Conditioning in Mid and Large Size Tokamaks

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In the presence of high permanent magnetic fields, as it will be the case in ITER, conventional DC glow discharges can not be used anymore. This paper reports on the recent assessment of the Ion Cyclotron Wall Conditioning (ICWC) technique for isotopic ratio control, fuel removal and recovery after disruptions, which has been performed on TORE

SUPRA (all-C), TEXTOR (all-C), AUG (all-W) and JET (Be/C).

ICWC discharges were produced using the standard ICRF heating antennas of each device, at different frequencies and toroidal fields, either in continuous or pulsed mode. Special attention has been paid to the careful diagnosis of these low temperature, partially ionized plasmas. The conditioning efficiency was assessed from the flux of desorbed and retained species, measured by means of mass spectrometry.

ICWC scenarios have been developed in D or H plasmas for isotopic exchange. The H (or D) outgassing was found to increase with the D (resp. H) percentage in the mixture. The isotopic ratio, measured during ohmic shots was significantly increased, with wall retention on the average two to ten times higher than desorption. In all devices, hydrogen isotope exchange is first found to increase shot after shot. Some signs of equilibrium between the walls and the RF discharge are only observed on TEXTOR and TORE SUPRA after 3 min and 10 min of ICWC, respectively. The flux of high energy (up to 50 keV) CX neutrals, measured on JET with a neutral particle analyzer, was found to be to low to explain retention nor desorption quantitatively.

Average fuel removal rates ranging from 10^{16} $\mathrm{Dm^{-2}{\cdot}s^{-1}}$ to 10^{18} $\mathrm{Dm^{-2}{\cdot}s^{-1}}$ have been achieved in Helium ICWC discharges. Whereas no He retention could be observed on all-C devices, a large fraction of the injected He was found to be retained in JET and AUG. In the latter case, however, this observation is in agreement with the observed He storage in W materials by He glow discharges in laboratory experiments, and the high He plasma impurity concentrations reported in AUG.

He-ICWC discharges have been successfully used on TORE SUPRA to recover normal operation after disruptions, when subsequent plasma initiation would not have been possible without conditioning. Analysis of the released gas favours ICWC operation in pulsed mode, which minimizes re-ionization and re-deposition of wall-desorbed species.

FTP/P1-27

On First Wall and Dust Issues in Fusion Devices

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The first wall plays crucial role in the design of any fusion reactor. However, the physics of plasma-wall interactions in fusion devices resulting in the modification of surface morphology and wall material properties, formation of co-deposited layers, bubbles, “fuzz”, dust, etc. is still poorly understood and in many cases there are no even rudimentary models explaining already observed phenomena. Here we address some issues related to the physics of the first wall and dust in fusion devices: The formation of “hot spots” related to thermal bifurcation caused by the synergy of wall evaporation and plasma transport; We present newly developed model of “fuzz” growth, which explains all major experimental observations; We report on recent progress in the modeling of hydrogen retention in first wall material with WALLPSI code, and the impact plasma recycling, which can be altered of hydrogen adsorption by Li wall, on edge plasma performance. We also consider the physics of dust-grain interactions for both non-spherical and large grains and discuss the synergetic effects of dust impact on edge plasma parameters and dust dynamics found with self-consistent dust and edge plasma modeling performed with coupled dust (DUSTT) and edge plasma (UEDGE) transport codes.

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FTP/P1-28

Consequences of Fatigue on Heat Flux Removal Capabilities of W Actively Cooled Plasma Facing Components

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Extensive R&D programmes have been performed in Europe to develop reliable actively cooled plasma facing components (PFCs) for the next fusion experiment like ITER. These activities focus on the development and fabrication of new plasma facing materials in terms of compatibility with plasma wall interaction and plasma scenarios. Key issues related to intense heat loads, hydrogen trapping, impurity generation from overheating surface and heat removal capability up to 20 MW/m^2 in steady-state conditions are as many challenges in the development of high performing PFCs. Wear resistant armour materials are foreseen to face the plasma, with low tritium retention property and intimate bonding to cooled structures. Within this framework, the tungsten (W) is increasingly considered as a prime candidate armour material facing the plasma in tokamaks. However, this material has not been yet used intensively in tokamaks and effect of fatigue on its long term behaviour is still rather unknown under operation.

Existing fusion devices do not provide yet the conditions required to assess actively cooled PFCs exposed to stationary thermal loads up to 20 MW/m^2 and sufficiently large cycle numbers ($>$ 1000 cycles). Hence, high heat flux tests, using electron beam, have been performed to assess the fatigue life-time of different bonding techniques as well as to validate design concepts as regards actively cooled W armoured plasma-facing components.

In this paper recent results are discussed in terms of heat removal capability and thermal fatigue performances at high heat flux for various types of actively cooled prototypes with W armour, including most recent developments.

First results showed promising behaviour in terms of heat flux removal capability up to 10 MW/m^2 but the bonding to cooled structure and the embrittlement of W armour materials are still considered unfavourable regarding high temperature deformation and cyclic fatigue for heat fluxes higher than 10 MW/m^2.

Therefore, the various promising developments which are in progress to reduce these drawbacks and to meet the operation demand of high power and long pulse in the future fusion reactor devices will be also discussed in this paper.

FTP/P1-29

Design and Development of Lower Divertor for JT-60SA

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Lower single null closed divertor with vertical target will be installed at the start of the experiment phase for JT-60 Super Advanced (JT-60SA). Reproducibility of brazed CFC (carbon fiber composite) monoblock targets for a divertor target has been significantly improved by precise control of tolerances and metallization inside CFC blocks. Divertor cassette with fully water cooled plasma facing components and remote handling (RH) system shall be employed to allow long pulse high performance discharges with large neutron yield and they are designed compatible with limited position and size of maintenance ports. Static structural analysis for dead weight, coolant pressure and electromagnetic forces shows that displacement and stress in the divertor module are generally small.

FTP/P1-30

Fully Actively-cooled in-vessel Components of EAST Tokamak

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EAST is the first full superconducting tokamak equipped to actively cooled plasma facing components (PFCs) which design and assembly were firstly finished in May 2008. During the two experiment campaigns in 2008 and 2009, the in-vessel components with full graphite tiles as first wall had been operated successfully. However, in some cases, fast particles have been observed, which locally increased power flux density, leading to damage of some of PFCs and other in-vessel components, and also for the reason of electromagnetic force. In view of safety operation with actively-cooled PFCs for handling of large input power over long pulse discharge, some R&D and maintenance were accomplished to improve the in-vessel components are presented here. For the purpose of large plasma current (1 MA) operation, the previous separate up and down passive stabilizers in low field were proposed to be electrical connected to stabilize plasma in the case of vertical displace events (VDEs). The design and activation work are described in this paper.

FTP/P1-31

Plasma Facing Material Selection: A Critical Issue for Magnetic Fusion Power Development

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This paper proposes a possible development approach that may satisfy the requirements for plasma facing materials (PFM). Five fundamental requirements will have to be satisfied for the selection of suitable PFM for the next advanced DT device beyond ITER: heat flux removal, acceptable lifetime, acceptable contamination to the plasma core, radiation resistance and the ability to withstand occasional rapid transient events like disruptions. If the materials selected for ITER were to be used in an advanced DT device, C and Be tiles would suffer radiation damage, and W could also suffer damage from neutrons and helium ions. Li has been proposed as a possible PFM option. However, due to vaporization and subsequent transport into the plasma core, modeling indicates that Li will have a maximum allowable surface temperature of $< 400°C$. Because of the required minimum allowed structural material temperature of $> 350°C$ for the ferritic steel substrate, the allowable surface heat flux removal capability of Li would be too low to make Li a credible surface material. For the projection of advanced solid PFM development, it is difficult to foresee any low activation metallic alloy that can outperform W-alloy. Ceramic surface materials tend to be brittle, possess low thermal conductivity and exhibit relatively high physical sputtering rates. An alternate approach for the PFM development would be the revival of real time boronization or siliconization research, such that the eroded material could be replenished. At the same time, an innovative approach like low-Z loaded W plate will be needed for the implantation of B or Si to about 10 micron equivalent thickness, such that the surface can withstand an occasional disruption without damaging or melting the underlying metallic substrate. From physics considerations and the need to control and mitigate rapid transient events like disruptions, Type-I ELMs and runaway charged particles, additional efforts will be needed to assure the uniformity of heat and particle flux distributions to the chamber wall and to maintain in real time a low-Z material coating on the chamber surface for steady-state operation.

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FTP/P1-32

On the Spherical Dusts in Fusion Devices

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It is now well known that nano- to micron size dusts are present in tokamaks [1, 2]. They are created by several creation mechanisms such as flaking, brittle destruction, arcing, and volume polymerization [3, 4]. Dusts created by flaking and brittle destruction have non-spherical shape while that created by arcing and volume polymerization have spherical shape: Spherical particles are spontaneously formed when critical density of molecules of materials exists in unit volume of plasma, so called "supersaturated vapor condensation". High, localized current during unipolar arcs in tokamaks leads to the local erosion of PFCs in a very short time interval which produces metal droplets. Although its creation mechanism and birth place are still under discussion, the presence of hydrogeneted carbon (a-C:H) nanoparticles of size smaller than 0.1 μm in diameter with a well defined Gaussian size disribution within a certain full-width half maximum (FWHM) is identified in many machines equipped with graphite plasma facing components (PFCs). This is a clear evidence of plasma volume polymerization [4] in fusion devices. This reveals that a parameter window for nanoparticle formation exists, even in fusion plasmas, with sufficient amount of hydrocarbons exceeding the critical density for the initiation of the process. Other dust creation processes like arcing, brittle destruction, or flaking of redeposited layers never give such a particle size distribution with spherical shape.

Existence of the metal droplets and a-C:H nanoparticles in fusion devices arises two interesting questions for future fusion devices like ITER: operational and safety issues. In this paper, we will describe the impact of metal droplets and a-C:H nanoparticles on the operation and safety of the fusion devices.

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FTP/P1-33

Generation and Characterization of High Heat-Flux Plasma-Flow for Divertor Simulation Studies using a Large Tandem Mirror Device

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This paper describes the results of high heat and particle flux generation performed at an end-mirror exit of a large tandem mirror device for divertor simulation studies. In a large tandem mirror device GAMMA 10 of Plasma Research Center, University of Tsukuba, a number of plasma production/heating systems with the same scale of fusion devices have been equipped and high-temperature plasmas have been produced. As a future research plan, making use of the advantage of open magnetic field configuration, we are planning to start a study of divertor simulation under the closely resemble to actual fusion plasma circumstances and to directly contribute the solution for realizing the divertor in ITER. For the above purpose, development of a new axi-symmetric divertor configuration like a toroidal divertor and high heat-flux experiment in the end-mirror region are started. In typical hot-ion-mode plasmas ($n_e \sim 2 \times 10^{18}$ cm^{-3}, $T_i \sim 5$ keV in the central-cell), the heat-flux density of 0.6 MW/m^2 and the particle-flux density of 10^{22} H/s$\cdot m^2$ were achieved in the case of only ICRF heating. Furthermore, overlaying of 300 kW ECH to the plasma attained a peak value of the net heat-flux up to 8 MW/m^2 on axis. The net heat-flux continues to increase with the ECH power and is expected to be achieved up to the level of 10 MW/m^2 within the available power of ECH, which almost corresponds to the heat load of the divertor plate of ITER. Above results give a clear prospect of generating the required heat-flux density for divertor studies by building up the plasma heating systems to the end-mirror cell. Detailed characteristics of obtained heat and particle flux are also discussed.

FTP/P1-34

Development of Liquid Lithium Limiter for Stellarotor TJ-II

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A new step in an improvement of TJ-II Heliac plasma performance is a development of two mobile poloidal liquid lithium limiters (LLL) allowing further progress in achievements of enhanced energy confinement owing to effective impurity and particle control. Main parameters, structural scheme and materials of LLL are presented and discussed. The possibility of LLL positioning relative to last closed magnetic surface, gas puffing through limiter, biasing and equipment with Longmuir probes and thermocouples, electrical heaters are ensure wide operation window. Study of gas release from lithium surface is stipulated in special volume included to the LLL structure. An analysis of LLL behavior in TJ-II plasmas produced with 400 kW ECRH power is presented and demonstrates possibility of its long term and flexible operation.

FTP/P6 — Poster Session

FTP/P6-01

Status of Project of Engineering-Physical Tokamak

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Main goals and preliminary parameters of Engineering-Physical Tokamak Project (EPT) are presented. EPT is medium size tokamak with warm magnetic system and aspect ratio in the range $2-3$, major radius $1.2-1.3$ m, toroidal field 2 T, plasma current 2 MA, elongation $1.7-1.9$, auxiliary heating power $10-15$ MW, pulse duration $5-10$ s and plasma in SN divertor configuration. It is assumed investigation of steady-state regime of plasma with parameters necessary for tokamak neutron source (TNS). EPT is seems as peculiar bridge from KTM tokamak to TNS.

FTP/P6-02

Importance of Helical Pitch Parameter in LHD-Type Heliotron Reactor Designs

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A system design code for heliotron reactors has been developed. Through the design window analysis by parametric scans over a wide design space, the importance of helical pitch parameter on LHD-type heliotron reactor designs has been clarified from the comprehensive standpoint as a reactor system. In the design studies of LHD-type heliotron reactors, one of the key issues is to secure sufficient blanket spaces. Then helical pitch parameter γ is quite important because it significantly affects both the coil and plasma shapes. In this study, parametric scans were carried out with 3 cases of γ: 1.15, 1.20 and 1.25 and the design window for fusion power of around 3 GW and density below 1.5 times the Sudo density limit were investigated. According to the past design studies, a blanket space of around 1.0 m is required. The achievable maximum value of the stored magnetic energy of helical coil system based on the ITER technological basis is considered to be 160 GJ. On the other hand, neutron wall loading needs to be kept below 1.5 MW/m^2 to adopt the long-life blanket concept. If helical coil major radius R_c are kept, blanket space increases

as γ decreases. To achieve 1.0 m blanket space, R_c of 16.2 m is required for $\gamma = 1.15$, while 17 m is required for $\gamma = 1.20$. There is no design point for $\gamma = 1.25$ which meets blanket space > 1.0 m and stored magnetic energy < 160 GJ. Since Sudo density limit is proportional to the square root of the ratio of magnetic field strength to plasma volume, the design window for magnetic field strength is widened as reactor size decreases. Then stored magnetic energy can be reduced with the progress in plasma confinement property in case of smaller γ. However, in case of $\gamma = 1.15$ there is few design points with large R_c (> 18 m) that meets neutron wall loading of < 1.5 MW/m^2. On the other hand, there is a design window that meets all engineering constraints with $R_c > 17$ m in case of $\gamma = 1.20$. However, neutron wall loading of 1.5 MW/m^2 is not a crucial constraint and there is a possible design window for $\gamma = 1.15$. Considering a small fusion power (i.e., 2 GW), there is a large design window that meets all engineering constraints in case of $\gamma = 1.15$. Consequently, helical pitch parameter needs to be optimized with a design concept.

FTP/P6-03

Conceptual Design Study of Superconducting Spherical Tokamak Reactor with a Self-Consistent System Analysis Code

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Spherical tokamak (ST) plasma has the potential of high beta operation with high bootstrap current fractions. Recent progress in the development of superconducting material, promising much higher engineering critical current density of 10 kA/cm^2 for high magnetic fields beyond 20 T by operating below liquid nitrogen temperature, led us to investigate the possibility of employing the superconducting TF coil in the aspect ratios of $1.5 \sim 2.5$. In the ST reactor, the radial build of TF coil and the shield play a key role in determining the size of a reactor. For self-consistent determination of these components and the physics parameters, the system analysis code is coupled with the one-dimensional radiation transport code, ANISN. As a reference design the ST reactor with outboard blanket only is considered where tritium self-sufficiency is possible by using an inboard neutron reflector instead of breeding blanket. In addition to metal hydride as an improved shielding material, high temperature superconducting magnets with high critical current density are used to open up the possibility of a fusion power plant with compact size and smaller auxiliary heating power simultaneously at low aspect ratio. Extending to high magnetic field regime with HTSC may lead to more stable operation regimes in ST reactor with high safety factor by lowering plasma current. With the use of advanced technology in the shield and superconducting TF coil, conceptual design of a compact superconducting ST reactor with aspect ratio of less than 2 will be presented as a viable power plant.

FTP/P6-04

Research Regimes and Design Optimization of JT-60SA Device towards ITER and DEMO

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The JT-60SA has been designed as a highly shaped large superconducting tokamak with variety of plasma actuators in order to satisfy the central research needs for ITER and DEMO. JT-60SA is capable of confining break-even-class deuterium plasmas lasting for a duration (typically 100 s) longer than any timescales characterizing the elementary plasma processes, such as current diffusion and particle recycling. JT-60SA also pursues full non-inductive steady-state operations at high values of the plasma pressure exceeding the no-wall ideal MHD stability limits. JT-60SA produces a wide range of plasma equilibrium covering a DEMO-equivalent high plasma shape parameter $S \sim 7$ with a low aspect ratio $A \sim 2.5$ at the maximum plasma current $I_p = 5.5$ MA and ITER-shape. JT-60SA has a strong heating power (41 MW) allowing variety of heating, current drive, and momentum input combinations using co-tangential, counter-tangential, and perpendicular NBs (24 MW/85 keV), the negative ion source co-tangential NB (10 MW/500 keV), and 7 MW/110 GHz ECRF. High beta plasmas are stabilized with the stabilizing shell, the resistive wall mode (RWM) stabilizing coils, and the error field correction coils. The water-cooled CFC monoblock divertor targets are compatible with the heat flux of 15 MW/m^2. The W-shaped divertor with a V-corner enhances divertor radiation and the divertor pumping speed can be changed by 8 steps up to 100 m^3/s.

With these capabilities, JT-60SA enables explorations in ITER- and DEMO-relevant plasma regimes in terms of the non-dimensional parameters (beta, the normalized poloidal gyro radius, and the normalized collisionality) under ITER- and DEMO-relevant heating conditions (such as dominant electron heating and low central fueling, and low external torque input). Under this condition, heat/particle/momentum transport, L-H transition, ELM/RMP/Grassy-ELM characteristics, the pedestal structure, high energy ion behaviors and the divertor plasma controllability are quantified. By integrating these studies, the project proceeds “simultaneous and steady-state sustainment of the key performances required for DEMO” with integrated control scenario development. Assuming $HH = 1.3$, the expected I_p for high β_N (= 4.3), high bootstrap fraction (= 66%) full non-inductive current drive is 2.3 MA at the Greenwald density ratio (= 1).

FTP/P6-05

Current Status and Facility Operation for KSTAR

Na, H.K.[1], Chu, Y.[1], Kim, K.P.[1], Park, M.K.[1], Park, D.S.[1], Kim, Y.O.[1], Park, K.R.[1], Cho, K.W.[1], Kim, J.S.[1], Park, S.H.[1], Yonekawa, H.[1], Woo, I.S.[1], Han, W.S.[1], Hong, J.S.[1], Baek, S.[1], Lee, T.G.[1], Park, J.S.[1], Lee, S.I.[1], Lee, W.R.[1], Park, J.S.[1], Kim, M.K.[1], Joo, J.J.[1], Moon, K.M.[1], Sajjad, S.[1], Song, K.H.[1], Park, J.M.[1], Kim, H.S.[1], Kwon, M.[1], Lee, G.S.[1], and KSTAR Team[1]

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KSTAR began high magnetic field (Toroidal Field: 3.5 T) and reliable performance plasma operation, which required implementation of safe and stable KSTAR operation including superconducting magnets. Reliable tokamak operation was conducted to achieve the goal for the 2010 KSTAR campaign to produce D-shaped plasma with the targeted current of 500 kA maintained for 5 seconds in duration. For conditioning of the vacuum vessel environment, it was expected that the 2010 KSTAR campaign includes a PFC baking operation with a designed maximum temperature of 350°C to improve the wall condition of the vacuum vessel. The Tokamak Monitoring System (TMS) for the cryogenic and superconducting magnet system and the Quench Detection System (QDS) for the detection of quench events were carefully inspected. The final goal of operating the KSTAR Integrated Control System (KICS) is to perform the roles of tokamak operation and plasma experiments with sustained stability, higher availability and security. Control systems were upgraded and added in order to provide a reliable control environment, and so they could contribute to achieving the goal of reliable performance plasma and supporting international collaboration work. The cryoplant is one of the most important systems among KSTAR's utilities because it supplies liquid helium to the superconducting magnet, which requires continuous operational capabilities of maintaining a cool-down of the system. The faults during the campaign of 2010 were analyzed for hardware, software and interlock systems in order to reduce operation errors during future campaigns and to protect the machine including the superconducting magnets. The analysis also showed what kinds of spare components were needed to prepare for emergency situation. Systematic KSTAR operational procedure is also required in order to unify various individual systems.

FTP/P6-06

Progress toward Attractive Stellarators

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Stellarators offer robust physics solutions for overcoming major challenges facing MFE. Stellarator research now increasingly focuses on engineering goals — reducing the technical risks in the construction and maintenance of future large-scale stellarators. Using the ARIES CS design as a starting point, compact stellarator designs with improved maintenance characteristics have been developed. The research focuses on approaches to enable a sector-maintenance approach, as envisioned for example in ARIES AT, by making the outer coil legs nearly straight and parallel. Aspect-ratio scans for optimized high-beta quasi-axisymmetric stellarators show that the toroidal excursions in the modular coils decrease with increasing aspect ratio such that planar parting surfaces appear, and radial removal of large sectors becomes possible. An alternative to increasing the aspect ratio, which has the disadvantage of leading to larger reactor size, is to make a qualitative change in the technology. In particular, magnetics studies show that bulk high-temperature superconducting (HTS) monoliths can be used to help shape the magnetic field and simplify the magnet coil geometry. Using the diamagnetic properties of bulk HTS material operating at elevated temperatures (> 10K), high-field, cryo-stable, highly complex magnet field topologies can be generated by arranging HTS tiles on a shaped, segmented internal support structure that can be removed radially through large port openings. Straight lines, flat surfaces, and simple curves are used wherever possible in the structure designs to improve manufacturing feasibility. With the superconducting tiles being used to shape the magnetic field, the main field coils can be planar and relatively few in number (for example, six), allowing access for removal of large sectors.

FTP/P6-07

A DT Neutron Source for Fusion Materials Development

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Fusion energy will require materials to withstand the harsh bombardment of energetic fusion neutrons and plasmas. The 2 MW Deuterium-Tritium Dynamic-Trap Neutron Source (DTNS) concept described here would supply a 14 MeV neutron flux over an area of a square meter for material and subcomponent evaluation. DTNS is based on the Novosibirsk Gas Dynamic Trap (GDT). Neutrons are produced by energetic ions trapped between axisymmetric magnetic mirrors. The energetic ions are surrounded and imbedded in warm plasma that provides both macro- and micro-stability. The recent GDT achievement of 60% beta now provides a basis for a dimensionless extrapolation to a DTNS design, of the same size as GDT, by increasing the neutral beam energy to 80 keV and the magnetic field to 1 Tesla. This paper describes the features of DTNS, physics based scaling methodology, engineering design concepts, methods for material evaluation, and tritium processing requirements.

FTP/P6-08

Tokamak KTM Progress Activity for Preparation on First Plasma Start-up

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The activities are carried out in the Republic of Kazakhstan on creation of experimental complex on the basis of spherical tokamak KTM. The complex is meant for study of structural materials and components of future fusion reactors. At present, the following works have been completed: vacuum chamber was placed at the tokamak support; the vacuum chamber's components (excluding graphite tiles) were mounted, namely, movable divertor device (MDD), passive stabilization coils, transport-sluice device (TSD), and equatorial nozzle plugs. The electromagnetic system has been mounted: poloidal and toroidal coils were mounted at proper places and fastened; central solenoid and central column were mounted. Power structure has been mounted; system for control of vacuum chamber temperature was mounted and preliminary adjusted. Mounting, adjustment and preliminary pumping of vacuum system was completed, vacuum of 10^{-6} torr was reached without preliminary annealing of vacuum chamber and installation of graphite tiles. Partial mounting of water cooling system and pulse power supply system for the

electromagnetic system of tokamak KTM was carried out. The diagnostics of first order of the tokamak KTM were adjusted.

After completion of mounting of all the systems, start-up activities for all the systems will be carried out. Trial start of tokamak KTM with realization of disruption stage is scheduled on 2010. Physical start-up of tokamak KTM will be carried out in 2011.

FTP/P6-09

Status and Result of the KSTAR Upgrade for the 2010's Campaign

Yang, H.L.[1], Kim, Y.S.[1], Park, Y.M.[1], Bae, Y.S.[1], Kim, H.K.[1], Kim, K.M.[1], Lee, K.S.[1], Kim, H.T.[1], Bang, E.N.[1], Joung, M.[1], Kim, J.S.[1], Han, W.S.[1], Park, S.I.[2], Jeong, J.H.[2], Do, H.J.[2], Lee, H.J.[1], Kwag, S.W.[1], Chang, Y.B.[1], Song, N.H.[1], Choi, J.H.[1], Lee, D.K.[1], Kim, C.H.[1], Jin, J.K.[1], Kong, J.D.[1], Hong, S.L.[1], Park, H.T.[1], Kim, Y.J.[1], Kim, S.T.[1], Im, D.S.[1], Joung, N.Y.[1], Namkung, W.[2], Cho, M.H.[2], Kwak, J.G.[3], Joung, S.H.[3], Jin, J.T.[3], In, S.R.[3], Wang, S.J.[3], Kim, S.H.[3], Kwon, M.[1], and KSTAR Team

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The Korea Superconducting Tokamak Advanced Research (KSTAR) is under drastic changes and upgrades for preparation of the 2010's campaign. The Plasma Facing Components (PFCs) such as limiters, divertor, and passive plate are to be fully installed to meet the requirement of the D-shaped, diverted plasma. The segmented In-Vessel Control Coil (IVCC) system was successfully installed in the vacuum vessel, and is being electrically connected to form a circular coil for Internal Vertical Control (IVC) in the 2010's campaign. The In-Vessel Cryopump (IVCP) was also installed prior to installation of the PFCs.

Construction of the first neutral beam injection (NBI) system is in progress, and the first NBI will demonstrate the first neutral beam injection into the KSTAR tokamak plasma as a course of system commissioning. Several important heating systems including ICRF, LHCD, and ECCD are being upgraded and developed for the 2010's and 2011's campaign. The Magnet Power Supply (MPS) system is also updated for the fully independent operation of lower and upper PF coils.

This paper will report the key features, status, result of the KSTAR upgrade in detail.

FTP/P6-10

Key Physics Issues of a Compact Tokamak Fusion Neutron Source

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The fast track to fusion energy applications is linked with the development of intense fusion neutron sources (FNS), which are of interest for many branches of technology and basic research, e.g. material science, neutron scattering, transmutation of elements, production of medical isotopes and nuclear fuel. The compact spherical tokamak is considered as an efficient device for neutron production. A MW range level of D-T fusion reaction could be realized on the basis of contemporary fusion technologies in a tokamak with major radius as small as 0.5 m at magnetic field ~ 1.5 T, heating power less than 15 MW and plasma current $2-3$ MA. The problem of steady state operation (SSO) remains critical for the device, and very thorough optimization of physics and technology is needed to get desirable performance. In this report we discuss key physics issues of SSO in a compact tokamak FNS. The basic motivation for the conceptual analysis of this device is reduction of the neutron load on the first wall with capital cost and operation expenses substantially below those of Component Test Facility (CTF) designs, while retaining the capability to produce neutron fluence comparable to that of a contemporary accelerator neutron source, or even higher. The major problems are plasma start-up, sustaining long pulse, and exhaust. Additional issues are the stability, plasma facing components, fuel cycle and the device maintenance. MHD stability has been analyzed using SPIDER, DINA and KINX codes. In such small and high density device, the contribution of neutrons produced by the beam-plasma interaction is dominating, thus we need a special analysis of fast ion losses. Evaluations using ASTRA, DINA, NUBEAM codes and semi-analytic models have shown that the performance is very sensitive to the confinement enhancement factor and the energy of the neutral beam injection. The modeling shows that the off-axis NBI current drive together with bootstrap current may be sufficient for steady state operation. Important issues of the device are the thermal wall load and the power load to the divertor of the order or even higher than ITER values. The analysis of the device neutronics has been performed using MCNP and IAEA FENDL. Our conceptual design studies demonstrate feasibility of the Compact Tokamak Fusion Neutron Source based on a small spherical tokamak.

FTP/P6-11

Objectives, Physics Requirements and Conceptual Design of an ECRH System for JET

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The future JET programme, after the installation of the ITER-like wall, will be mainly focused on the consolidation of the physics basis of the three main ITER scenarios, i.e., the ELMy H-mode, the hybrid scenario and the advanced steady-state scenario. In ITER, these scenarios will make substantial use of Electron Cyclotron (EC) waves, for heating as well as for control of both the MHD activity and the current density profile. Therefore, a programme for preparation, validation and optimization of the ITER scenarios in present tokamaks would strongly benefit, and even require, an ECRH/ECCD system. This gives a strong motivation for examining the feasibility of the construction and implementation of such a system in JET, for an intensive exploitation before the start of ITER.

This paper reports on the results of feasibility studies performed by a EU-Russia project Team for a JET ECRH system. It will be organised in four main parts:

- Objectives and rationale for the implementation on an ECRH system on JET
- Physics studies, aiming at a definition of the basic parameters of the system (wave frequency, launching geometry, power required)
- Exploration of the main technical options (wave sources, transmission line and windows, antenna design, power supply, auxiliaries, layout and port allocation, control systems, diagnostics)
- Cost, time schedule, manpower and risk analysis

Work supported by EURATOM and carried out under EFDA.

FTP/P6-12

Development of Over-1 MW Gyrotrons for the LHD and the GAMMA 10 ECH Systems

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The Electron Cyclotron Heating (ECH) is one of the key tools of the plasma heating, current drive, plasma control for the magnetic confinement fusion devices. For the ECH upgrade program of LHD and GAMMA10, over-1 MW power gyrotrons with TE18,6 cavity and a diamond window at 77 GHz, and with TE8, 3 cavity at 28 GHz have been developed for LHD and GAMMA 10 in the joint program of NIFS and University of Tsukuba. The maximum outputs of 1.5 MW for 1.6 s at 77 GHz and $\sim$ 1 MW at 28 GHz were obtained, which are the new records in these frequency ranges. The results of 1.6 MW for 0.5 s, 1 MW for 5 s, 300 kW for 7 min and 200 kW for 21 min were also achieved at 77 GHz. In the long pulse operation, it is found that the stray RF is the major cause limiting the pulse length. Design improvements of the diffraction loss, the cavity and alpha dispersion have made the 77 GHz tube performance better, as demonstrated 1.5 MW output. The 77 GHz gyrotrons have already been installed in the LHD ECH system and more than 3 MW has been injected into LHD plasma.

FTP/P6-13

Progress of High Power and Long Pulse ECRF System Development in JT-60

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Gyrotron output power of 1.5 MW with the pulse duration of 4 s was recorded on the JT-60 ECRF system by developing a new operation technique. In order to achieve high power oscillation of 1.5 MW without troubles on the collector, the heat load of the collector is to be less than 1.7 MW, which was estimated by the collector temperature measurements on high power and long pulse oscillations in JT-60. It corresponds to the efficiency of $\sim$ 45% while the heat load of the conventional operation with fixed operation parameters was higher than 2.1 MW (efficiency of less than 40%) at long pulse. The electron pitch factor was optimized just after the gyrotron operation starts ($\sim$ 100 ms) by active control of the anode voltage. It led to reduction of the collector heat load by 20% with respect to the conventional operation (without active control) for long pulse operation by enhancement of the efficiency. The heat load of the collector at 1.5 MW operations was acceptable for steady state operation. The advantage of this technique is to enable a quick access to the hard-excitation region, where the oscillation cannot be excited from the noise level but

once the oscillation is established the high efficiency operation is possible, which is effective when the quick RF injection is required such as for the application to the neoclassical tearing mode stabilization. Modification of power supply, transmission line and control system, which enables pulse duration of 50 s (formerly 5 s), has been made in order to clarify the performance of the ECRF system in long-pulse operations. A gyrotron with an improved mode convertor was newly fabricated to reduce the stray radiation, which had so far hindered long-pulse operations through unacceptable heat load. A conditioning operation of the improved gyrotron is proceeding smoothly up to 17 s at 1 MW. It was also confirmed that the stray radiation on DC-break between body and collector electrodes was reduced to be acceptable for steady state operation, and consequently issues recognized as major problems for expanding the pulse length of the gyrotron oscillation were resolved. These progresses significantly contribute to enhancing the high power and long pulse capability of the ECRF system toward JT-60SA, where the total output power of 9 MW for 100 s is planned.

FTP/P6-14

Design and Commissioning of a Novel LHCD Launcher on Alcator C-Mod

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The goal of the lower hybrid current drive (LHCD) experiment on Alcator C-Mod is to demonstrate and study the full non-inductive high performance tokamak operation using parameters close to that envisioned for ITER in terms of LHCD frequency and magnetic field. Previously, up to 1.2 MW of net LHCD power at 4.6 GHz has been successfully launched for 0.5 s with a traditional grill launcher. To explore higher power and longer pulse regimes, a new LHCD launcher (LH_2) was designed and fabricated. The new launcher is based on a novel four way splitter concept which evenly splits the microwave power in the poloidal direction. Sixteen splitters are stacked in the toroidal direction, creating the 16 columns and 4 rows of active waveguides. A total of 8 passive waveguides (one on each side of each row) are installed to reduce the reflection on the edge columns. This design allows the simplification of feeding structure "jungle gym", while keeping the flexibility to vary the launched toroidal $N_\parallel$ spectrum from -3.8 to 3.8. To predict the LH_2 performance, an integrated modeling using TOPLHA and CST microwave studio was carried out, in which the antenna-plasma coupling problem and the vacuum side EM problem were solved self-consistently. Good plasma coupling over a wide range of edge densities and a clean $N_\parallel$ spectrum were confirmed. The poloidal variation of edge density was found to affect mainly the evenness of power splitting, suggesting the necessity of the forward and reflected power measurements at each row of the launcher. 16 dedicated sets of RF probes were installed on a carefully selected set of active and passive waveguides to measure the forward and reflected power in each of these waveguides. In addition, the LH_2 launcher is equipped with six Langmuir probes,

and three X-mode reflectometer waveguides to measure the density profile in front of the launcher. Experiments using LH_2 are scheduled to start in May. Initial results will be reported.

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FTP/P6-15

Development of a Plasma Current Ramp-up Technique for Spherical Tokamaks by the Lower-Hybrid Wave

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Spherical tokamaks (STs) have the advantage of superior stability at high beta, but to realize a compact fusion reactor, the central solenoid (CS) must be eliminated. The plasma current could be maintained mainly by the self-driven current during the steady-state burning phase, but there is no established method of non-inductive current ramp-up from zero to a high enough level required for fusion burn. The lower-hybrid wave (LHW) is commonly used in tokamaks, but this technique is believed to be unusable in ST plasmas with very high dielectric constant. Wave excitation, propagation and damping were evaluated by numerical modeling for the TST-2 spherical tokamak (R = 0.68 m, a = 0.25 m, B = 0.3 T, I = 100 kA). Wave excitation is calculated by the RF antenna simulation tool based on the finite element method (COMSOL). The plasma is modeled as a medium with cold plasma dielectric response and artificially enhanced loss. Excitation of a traveling fast wave by the TST-2 combline antenna was confirmed. The fast wave must be mode converted to the lower hybrid wave to achieve efficient current drive. The TORIC-LH full-wave solver, which can treat diffraction effects properly, was applied to the TST-2 LHCD experiment. It is shown that core current drive by LHW is possible in the low density, low current plasma formed by ECH. It is important to keep the density low during current ramp-up, and the wavenumber must be reduced as the current increases in order to maintain core current drive. The results of these calculations and the initial experiment on TST-2 will be used to design an optimized LHW antenna with appropriate polarization and wavenumber spectrum controllability for the current ramp-up experiment. Taking a conservative value for the current drive figure of merit, the steady-state driven current by 200 kW of RF power is estimated to be 150 kA, which should be adequate for evaluating the usefuleness of this technique up to a current level of 100 kA in TST-2. This technique should be usable during the low-density, low-current, non-burning start-up phase of a reactor to reach a sufficient current level needed for further heating. The success of the TST-2 experiment would provide a scientific basis for quantitatively evaluating the required CS capability for a low-aspect-ratio reactor.

FTP/P6-16

Commissioning Results of the KSTAR NBI System

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The neutral beam injection (NBI) system is designed to provide the ion heating and current drive for the high performance operation and long pulse operation of the Korean Superconducting Tokamak Advanced Research (KSTAR). KSTAR NBI consists of two beam lines. Each beam line contains three ion sources of which one ion source has been designed to deliver more than 2.5 MW of neutral beam power with maximum 120-keV beam energy for deuterium beam. Consequently, the final goal of the KSTAR NBI system aims to inject more than 14 MW of deuterium beam power with the two beam lines. According to the planned NBI system, the first NBI system is to demonstrate the beam injection into the KSTAR tokamak plasma in 2010's KSTAR campaign including the system commissioning of each components and subsystems.

The first ion source to be used for 2010's KSTAR campaign consists of the plasma generator, which was developed by Japan Atomic Energy Agency (JAEA) by means of KO-JA fusion collaboration agreement and whose validation test was completed at Korea Atomic Energy Research Institute (KAERI), and the KAERI accelerator. The ion source is mounted on the beam line system and connected with the high voltage cables and filament and arc powering cables coming from the high voltage deck.

This paper will present the overview of the KSTAR first beam line system and the results of the beam line commissioning and the first beam experimental results. Also, the future plan of the KSTAR NBI system development will be presented.

FTP/P6-17

Plasma Commissioning Scenario and Initial Tritium Inventory for Demo-CREST

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This paper discusses the plasma commissioning scenario and the relationship between the initial tritium inventory and the commissioning period for a demonstration reactor concept Demo-CREST. The tritium density ratio (T-ratio) control is applied to keep the high density operation preferable for the divertor heat-handling during gradual increase of the fusion power in the commissioning phase.

It is found that Demo-CREST can start from zero-fusion power operation with T-ratio 0%, in which the divertor heat-handling condition on the SOL density and the radiation power required for divertor heat load less than 10 MW/m^2 is similar to that of the ITER steady state operation. The operational space for Demo-CREST was investigated by the MHD stability code ERATO and the current drive analysis code DRIVER88. The divertor heat load condition is evaluated by the two point SOL-divertor model, and the initial operation point with T-ratio 0% is confirmed by the SOL-divertor transport code SOLPS5.0. An operation route keeping high density by the T-ratio control is also proposed for the commissioning. The critical issue on this commissioning scenario is the plasma confinement. In case of smaller T-ratio than 10%, a high plasma confinement $HH = 1.57$ similar to that of the ITER steady state operation is required.

This proposed operation route has a consistency with the start-up scenario without the initial tritium loading, in which tritium production is started from the DD fusion neutron by the beam direct fusion reaction. The relationship between the initial tritium inventory and the commissioning period is evaluated for Demo-CREST. When this startup scenario without initial tritium was applied to the operation route with the T-ratio control of Demo-CREST, the commissioning period (T_{com}) from T-ratio 0% to 50% results in $T_{com} =$ 360 days, under the condition of the total dead inventory 1000 g and the tritium breeding ratio (TBR) 1.05. When the commissioning period has to be shorten to $T_{com} = 90$ days, the required initial tritium increases up to about 1000 g. Decrease of TBR results in increase of initial tritium loading. When the commissioning period has to be shorten within $T_{com} = 90$ days, the tritium breeding during the commissioning phase is not effective to reduce the initial tritium loading even with the high TBR such as 1.10.

FTP/P6-18

Fusion-Biomass Hybrid Concept and its Implication in Fusion Development

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Limited Q plasma combined with fuel production from biomass is proposed to be possible option of fusion energy to be possible in the near future. High temperature blanket concept with LiPb and SiC, and conversion of cellulose and lignin to H_2-CO mixture at the temperature around 900°C at high efficiency over 90% were both experimentally tested by the authors. This product H_2-CO gas mixture can further yield either more hydrogen by Shift reaction, or artificial oil such as diesel by Fischer-Tropsch Synthesis, They cause little carbon dioxide emission, and the market is larger than electricity. By adding the original chemical energy of the waste biomass, total apparent energy conversion efficiency from fusion to the product as a form of gaseous fuel approaches 270%. Endothermic reaction is free from the thermal cycle efficiency limit, and combination with biomass energy drastically reduces energy multiplication requirement for plasma. This energy "multiplication" significantly relaxes the requirement of plasma Q factor, because net energy production by the plant is possible with $Q < 5$ plasma, compared with the requirement to exceed $Q > 20$ in the case of pure electricity generation from fusion. Unlike in the case of electricity generation, pulsed, and/or driven burning plasma may be used because the product is a fuel, that does not require steady state operation. This energy plant can be regarded as Fusion-Biomass hybrid, that is free from all the technical complication of the fission-fusion hybrid. Based on these features, the authors have designed a reduced scale tokamak of $Q \sim 5$ with $R \sim 5$ m, as a plant that can eventually demonstrate positive net energy production as fuel. Total thermal energy output is around 500 MW that is similar to ITER, and required power densities on in-vessel components are in the range to be possible and realistic ones those already applied to ITER, or with reasonable extension of the current technology. Exception may be the challenge of high temperature blanket, while small scale tests are already performed. This paper will also report our recent development of high temperature blanket, such as 900°C LiPb loop, SiC components, heat exchanger(IHX) and tritium recovery. This concept provides possible option of the early realization of fusion with opportunity of engineering maturity of reactor technology.

FTP/P6-19

The High Density Ignition in FFHR Helical Reactor by Neutral Beam Injection Heating

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The high-density up to 1.1×10^{21} m^{-3} has been observed in large helical device (LHD) pellet injection experiments. Although a high-density operation is advantageous to reduce the divertor heat flux via bremstrahlung radiations and to ease the pellet penetration, its thermally unstable control and heating of the high-density plasma have been major issues. The first issue to stabilize the thermal instability has been solved by the proportional-integral-derivative (PID) control algorithm based on the fusion power in the FFHR2m helical reactor. In this paper we propose the new operation scenario to solve the second issue using neutral beam heating for accessing the high-density ignition regime. In this scenario the density is kept at the low value less than $\sim 3 \times 10^{20}$ m^{-3} to ensure the NBI penetration and to pass the saddle point on POPCON to reduce the heating power at the same time. Above this density, high-density ignition is accessed by alpha heating after turning off neutral beam injection using the proposed PID control.

FTP/P6-20

Concept of Power Core Components of the SlimCS Fusion DEMO Reactor

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Engineering aspects of a compact fusion DEMO reactor SlimCS with water-cooled solid breeder blanket is presented. This paper mainly focuses on a novel concept of power core components, including blanket, conducting shell and high temperature shield. The concept will allow compatibility between high performance power core and engineering feasibility on water cooled reactors. It will be commonly applicable to any other water-cooled reactor being operated at a high beta exceeding no wall limit.

FTP/P6-21

Heat Flux Reduction by Helical Divertor Coils in the Heliotron Fusion Energy Reactor

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Based on the progress of high-density and high-temperature plasma experiments in the Large Helical Device (LHD), conceptual design studies on the heliotron-type fusion energy reactor FFHR are being conducted on both physics and engineering issues. In order to best utilize the built-in helical divertors in FFHR, we propose a new divertor sweeping scheme which could effectively reduce the divertor heat flux and mitigate the erosion of divertor plates. The concept employs a small set of helical coils, which we call "helical divertor coils". We find that the divertor legs can be moved effectively by modulating the amplitude of the current in the helical divertor coils by 1% of the amplitude in the main helical coils. It should be emphasized that despite the movement of divertor legs, the magnetic surfaces show almost no change with this scheme, which is different from the previously proposed divertor sweeping schemes for LHD. Inclining the divertor plates against divertor legs enlarges the width of strike points to $\sim$ 800 mm. If a fast sweeping is realized, the effective heat flux would thus be lower than 1 MW/m^2 on time average having the total power flow of $\sim$ 600 MW to the divertor regions with a 3 GW fusion power. If we employ 0.5 Hz frequency of sweeping, the required coil current would be doubled since the skin effect reduces the magnetic field to be half assuming a 0.4 m thick nuclear shielding in the vicinity of divertor areas. Erosion of divertor plates would also be mitigated even with quasi steady-state sweeping and the replacement cycle could be significantly prolonged. Regarding the engineering design of the helical divertor coils, we propose that these coils be fabricated using high-temperature superconductors represented by YBCO. The coils, consisting of 25 turns of 30 kA conductors, could be constructed with prefabricated segments and jointed on site. The losses in the coils generated at joints and by AC operations are of no serious concern with elevated temperatures at 60K or higher.

FTP/P6-22

The New JET Vertical Stabilization System with the Enhanced Radial Field Amplifier and its Relevance for ITER

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This paper describes the experimental activities carried out at JET for the integrated commissioning of the new enhanced radial field amplifier (ERFA) and the upgraded Vertical Stability control system. The main aim of this activity was not only the pure test of these new components but also the optimisation of their performance on plasmas.

This activity was based on the so-called model-based approach for the current, position and shape controller, which seems the most viable solution for ITER and future devices. The electromagnetic simulation models of the interaction between the plasma and the surrounding conductors was used for the design of the control system with a very limited activity of tuning of the large number of controller parameters during the experiments and a reduced risk of plasma disruptions.

One of the crucial issues was the determination of the most promising option amongst three different radial field turn configurations of the radial field circuit. The performance has been assessed on the basis of the response to controlled perturbations (vertical and divertor kicks) in a wide a range of configurations, so as to explore the system capability in response to controlled/reproducible perturbations as safely and time-efficiently as possible. A significant experimental time has also been allocated to the exploration of plasma scenarios with a large variety of edge localized modes (ELMs).

Many activities carried out at JET during this commissioning can be regarded as ITER relevant, and can therefore give significant indications for ITER controller design and any possible satellite experiment (like JT60-SA and FAST):

– assessment of controllers and power amplifiers designed using the model based approach;
– usage of in-vessel coils on the time scale of vertical stabilization;
– characterization of plasma response to the ELMs;
– response of magnetic diagnostics on the time scale of vertical stabilization.

FTP/P6-23

Feasibility of Graphite Reflector in Tritium Breeding Blanket

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Korea has proposed a Helium-Cooled Solid Breeder (HCSB) breeding blanket concept adopting graphite reflector for Test Blanket Module (TBM) program which aims to test and validate the concept relevant to DEMO and/or fusion power plants. It is unique feature among TBMs that the amount of beryllium is considerably reduced by replacing some of them with graphite satisfying the tritium self-sufficiency condition. However, there are still concerns on the graphite reflector related to the nuclear performance as well as material-related issues. It should be shown that tritium breeding capability is improved by adopting graphite reflector in fusion reactor conditions. The neutron shielding capability of a combination of the breeder, multiplier and reflector should be also compatible with a combination of breeder and multiplier only. If it is not, the radial build should be increased or more power is required to drive the helium refrigeration system for the superconducting magnets, both are not desirable. In the present study, a nuclear analysis is performed under the fusion reactor condition to address the feasibility of graphite reflector in breeding blanket. While the breeding zone thickness is fixed, sensitive tests are performed with varying graphite amount, i.e. increasing graphite and decreasing beryllium or vice versa. Introducing an engineering factor, TBR/beryllium weight, optimal reflector thickness is obtained on the basis of reduction of beryllium amount and enhancement of tritium breeding capability.

FTP/P6-24

Progress on the Development of Fabrication Technology for the KO HCML TBM

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The Republic of Korea has proposed and designed a helium cooled molten lithium (HCML) test blanket module (TBM) to be tested in the ITER. Ferritic Martensitic Steel (FMS) and beryllium are designed to be used as structural material and armor for the TBM first wall (FW) in this design, respectively. In order to develop the fabrication method for the TBM FW, the joining methods of FMS to FMS and Be to FMS have been developed with hot istotatic pressing (HIP). For joining FMS to FMS, mock-ups were fabricated with an HIP (1050°C, 100 MPa, 2 hours). For joining Be to FMS, two mock-ups were fabricated with the same method (580°C, 100 MPa, 2 hours) using different interlayers. Then, in

order to evaluate the integrity of the fabricated mock-ups, they were tested at the high heat flux (HHF) test facilities, KoHLTs (Korea Heat Load Test) under 1.0 MW/m^2 and 0.5 MW/m^2 heat fluxes of up to 1000 cycles.

FTP/P6-25

Progress in R&D Efforts on Neutronics and Nuclear Data for Fusion Technology Applications

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The objective of this paper is to present the progress achieved over the past two years in the R&D programmes on neutronics and nuclear data for fusion technology applications. The focus is on the recent achievements in providing consistent high quality nuclear data evaluations including co-variance data, in developing advanced computational tools such as the McCad conversion software for the generation of Monte Carlo analysis models from CAD geometry data, and the MCsen code for efficient Monte Carlo based calculations of sensitivities/uncertainties of nuclear responses in arbitrary geometry, and in conducting neutronic benchmark experiments on breeder blanket mock-ups.

A consistent evaluation of neutron cross-section data was performed for the stable Cr isotopes up to 150 MeV. The evaluation is based on nuclear model calculations, experimental cross-section data and uncertainty information. The calculation of co-variances is based on the Unified Monte Carlo method and takes into account both experimental uncertainty information and nuclear model deficiencies. The McCad geometry conversion tool has been developed to enable the CAD geometry conversion for the Monte Carlo code MCNP. Recent enhancements include advanced algorithms for the generation of void cells, improved visualisation features, the integration of a simple material management system, and the extension for the Monte Carlo code TRIPLO-4. The MCsen Monte Carlo code for sensitivity/uncertainty calculations has been upgraded to enable the efficient calculation of sensitivities to nuclear cross sections of nuclides contained in different materials. Recent MCsen applications include sensitivity/uncertainty analyses of the benchmark experiment on a neutronics mock-up of the Helium Cooled Lithium Lead (HCLL) Test Blanket Module (TBM) as well as corresponding analyses on the HCLL TBM integrated into a Test Blanket port in ITER. The recent benchmark efforts were devoted to experiments on a mock-up of the HCLL TBM including measurements of the Tritium production applying different techniques, measurements of dedicated reaction rates using the activation foil technique, and measurements of fast neutron/ photon flux spectra and of time-of-arrival spectra of slow neutrons. Good agreement between the measurements and the MCNP based calculations was obtained for all experiments.

FTP/P6-27

Data Acquisition System for Steady State Experiments at Multi-Sites

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A high-performance data acquisition system (LABCOM system) has been developed for steady state fusion experiments in Large Helical Device (LHD). It also acquires experimental data of multiple remote machines through the 1 Gbps fusion-dedicated virtual private network (SNET) in Japan. The LABCOM system consists of distributed data acquisition (DAQ) computers, data storage devices, and the index database of data locations. Its key objectives are (1) real-time DAQ with the same sampling rates as burst acquisition, and (2) scalability of both the performance and quantity, even in the number of sites. LHD now uses 80 DAQs in parallel and acquires 10.6 GB/shot raw data in short-pulse experiments of 3 min. iteration. In steady state operation, huge kHz or MHz fluctuation data for plasma diagnostics must be processed in real time. In LHD, each DAQ computer can deal with max. 110 MB/s continuous data. The DAQ cluster has established the world record of acquired data amount of 90 GB/shot, which almost reaches the ITER data estimate of 100 or 1000 GB/shot. As for the data storage, the massively parallel processing (MPP) structure is important for scalable input/output performance and data redundancy. Hundreds of tera-byte compressed data are stored in the two-stage storage devices. The data mediation service is a distinguishing characteristic that manages the peer-to-peer data handling among many server and client computers. As every element of the LABCOM system are distributed on the local area network (LAN), the data of remote fusion devices are acquired simply by extending the LAN to the wide-area SNET. LABCOM system acquires data from three remote experiments of different universities. The speed lowering problem in distant TCP/IP communication, e.g. 60 Mbps on 1 Gbps SNET, is improved by using the optimized congestion control and packet pacing technology. Its bandwidth tests between Japan and France achieved effective 881 Mbps over 10 000 s. A light-weight access control is implemented in the LABCOM system because it provides enough security for a closed VPN like SNET. Toward the fusion goal, a common data-accessing platform is indispensable so that physicists can easily make detailed comparisons between multiple large and small experiments. The demonstrated bilateral scheme will be analogous to that of ITER and the supporting machines.

FTP/P6-28

Thermohydraulic Characteristics of KSTAR Magnet System using ITER-like Superconductors

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KSTAR (Korea Superconducting Tokamak Advanced Research) has been operated since 2008 and 3rd campaign will be carried out in 2010. In the world, KSTAR is the only superconducting tokamak adopting Nb3Sn so it is a very interesting item for not only KSTAR Team but also ITER people even though KSTAR cable-in-conduit conductor (CICC) is formed into rectangular shape and has no central hole for helium passage. Of course several preliminary tests of Nb3Sn coil and magnet like ITER TFMC and CSMC were carried out before KSTAR operation and the EAST has also operated since 2006, however only KSTAR can show us the integrated performance as full magnet system using ITER-like superconductors. We have already achieved maximum TF current of 36 kAand bipolar operation of PF coils during two KSTAR campaigns. The thermohydraulic characteristics and behavior like temperature and pressure variation of supplied helium according to the scenario, pressure drop inside cooling channels and the friction factor of CICC were measured and calculated. It is also possible to compare the physical phenomenon with the prediction by computational simulation after experiments.

In this paper, we introduce the thermohydraulic results of KSTAR magnet system up to now and mention the effect to ITER and other superconducting tokamak under construction.

FTP/P6-29

The JT-60SA Superconducting Magnetic System

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The JT-60SA experiment is one of the three projects to be undertaken in Japan as part of the Broader Approach Agreement, conducted jointly by Europe and Japan, and complementing the construction of ITER in Europe. The superconducting magnet system for JT-60SA consists of 18 Toroidal Field (TF) coils, a Central Solenoid (CS) and six

Equilibrium Field (EF) coils. Two conventional poloidal coils inside the vacuum vessel for the fast plasma position control. The TF generate the field to confine charged particles in the plasma, the CS provide the inductive flux to ramp up plasma current and contribute to plasma shaping, the EF coils provide the position equilibrium of plasma current and the plasma vertical stability. The TF coil case encloses the winding pack and is the main structural component of the magnet system. The six EF coils are attached to the TF coil cases through supports with flexible plates allowing radial displacements. The CS assembly is supported from the bottom of the TF coils through its pre-load structure. The design status of the JT-60SA superconducting magnetic system is reviewed the many challenges posed are described.

FTP/P6-30

Design Study of Plasma Control System on JT-60SA for High Beta Operation

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One of the important missions of JT-60SA is to demonstrate and develop steady-state high beta operation in order to supplement ITER toward DEMO. Specifications of plasma control system including the stabilizing plate, the resistive wall mode (RWM) control coils, the error field correction coils and fast position control coils were determined based on simulations and expected plasma regime was evaluated. Full-current-drive plasma with $\beta_{\mathrm{N}} = 4.3$ and $I_{\mathrm{p}} = 2.3$ MA will be achieved with these control systems. Study on Rigidity of RWM in the RFX-mod Reversed Field Pinch device is in progress for design finalization of the RWM control system. Simulation of plasma disruption was also performed to evaluate design values of Electro Magnetic forces of components.

FTP/P6-31

Development of Collective Thomson Scattering System using the Gyrotrons of sub-Tera Hz Region

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Collective Thomson scattering (CTS) is a unique and the most promising method to measure the ion velocity distribution function that is a key information in the reactor grade plasma. Recent developments of the high power gyrotron in the sub-Tera Hz region enabled us to envisage its application to CTS in the fusion relevant plasma parameters as in ITER. CTS in this frequency range has many advantages. First, the CTS condition is satisfied for large scattering angle larger than 90 degrees in the fusion grade plasma. This provides good spatial resolution. Second, the sub-THz wave does not suffer from refraction due to cutoff in plasma of the order of 10^{20} m^{-3}. Third, it is almost free from cyclotron absorption since its frequency is much higher than harmonics of the cyclotron frequency. Forth, then, the background ECE will be at a very low level. However, CTS in this frequency range has not been considered because of the lack of adequate power source. Taking account of the background ECE level, an output power of 100 kW would be required for the CTS from a high density plasma of the order of 10^{20} m^{-3}. For the first stage of development, we have fabricated a pulse gyrotron operating with 400 GHz frequency range at second harmonic using an 8T-superconducting magnet. The measured maximum power is 52 kW for TE6,5 mode and 37 kW for TE8,5 mode, respectively. These powers are new world records of second harmonic gyrotron in this frequency range. Moreover, another gyrotron operating at fundamental harmonic with 400 GHz frequency range is being designed. An output power of more than 200 kW is expected with this gyrotron. In parallel to the development of the sub-Tera Hz source, it is important to obtain the know-how of the CTS from the real plasma and to develop the method of receiving and analyzing the CTS spectrum. One of the 1 MW, 77 GHz gyrotrons and transmission/antenna system for that in LHD are selected as the probing source and probing/receiving system for the benchmark. A heterodyne receiver system is installed on the other end of the transmission line. After executing the receiver gain adjustments and calibration of each channel, clear CTS spectra are obtained. These data give us confidences that ECRH system in LHD works also well as CTS system and can be a good benchmark for the CTS using sub-THz gyrotron.

FTP/P6-32

Progress in Design and R&D for in-Vessel Coils for ITER, DIII-D, and JET

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Much progress has been made in the R&D and design of in-vessel coil systems for DIII-D, ITER, and JET as the result of a coordinated effort to develop acceptable performance, mechanical design, materials, analysis and manufacturing techniques for Edge Localized Mode (ELM) control. This paper presents the status of these design and R&D efforts at PPPL.

First wall and divertor erosion and damage caused by ELMs is a major hurdle on the route towards high performance, long pulse operation of ITER and achieving magnetic fusion in a reactor scale machine. Presently the most promising method of mitigating, or even completely suppressing ELMs is to apply resonant magnetic field perturbations (RMP) in the plasma edge. This technique to suppress ELMs was discovered on DIII-D, and experimentation continues on DIII-D and several other machines. Because of the importance of mitigating ELMs, a set of RMP coils is being designed for ITER, based on empirical criteria developed on DIII-D. A new set of center-post mounted coils, scheduled for installation in DIII-D beginning in 2010, has been designed to extend their physics studies A design study for JET has demonstrated the feasibility of a system of in-vessel RMP coils, including installation by remote handling, that would satisfy ELM suppression criteria for all of JET's ITER-relevant scenarios. Experiments with these upgrades will provide additional information towards the understanding of ELM control by RMP, and extend the dataset for extrapolation towards ITER-like plasmas.

There is some commonality of design aspects in the three designs, such as the need for high temperature insulation, high reliability due to their in-vessel locations, and the need for vacuum jacketing Prototypes of the DIII-D coils are being fabricated. This effort includes the development of low-distortion welding techniques for the vacuum jackets and qualification of high-temperature insulation, also required for JET. There are also differences in operational characteristics of the three tokamaks which have dictated significantly different design solutions. The in-vessel environments of the three machines leads to differences in material selection and in the need for remote handling. The primary design drivers for all 3 devices will be discussed in detail, as well as proposed design solutions.

FTP/P6-33

Experimental Studies of MHD Flow in a Rectangular Duct with FCIs

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The initiation and subsequent experiments are performed of the MHD flow in a duct with the flow channel insert (FCI). The four FCIs have differences in the conductivities (epoxy and 304 SS) and in the structures (pressure equilibrium holes, PEHs and pressure equilibrium slot, PES). The experimental results show that all of the FCIs result in the complex and drama velocity contribution in the cross section of the duct; which is out of the numeric simulation and classical MHD theory expectations due to the secondary flow MHD effect could not be involved. The experiments were done under the conditions: uniform transverse magnetic field $B \sim 2$ T, mean velocity $V \sim 6$ cm/s, the 304 SS rectangular duct 68×60 mm, FCI 54×46 mm, 3 mm in width of PES, 10 mm in diameter of PEH.

FTP/P6-34

Comprehensive First Mirror Test for ITER at JET with Carbon Walls

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Metallic mirrors will be essential components of all optical spectroscopy and imaging systems for plasma diagnosis that will be used on the next-step fusion experiment, ITER. Any change of the mirror reflectivity, will influence the quality and reliability of detected signals. On the request of the ITER Design Team, a First Mirror Test (FMT) has been carried out at JET during campaigns in 2005-2007 and 2008-2009. To date, it has been the most comprehensive test performed with a large number of test mirrors exposed in an environment containing both carbon and beryllium; the total plasma time (in 2005-2007 period) over 35 h including 27 h of X-point. 32 stainless steel and molybdenum (Mo-poly) flat-front and 45° angled mirrors were installed in separate channels of cassettes on the outer wall and in the Mk-II HD divertor: inner leg, outer leg and base plate under the load bearing tile. Post exposure studies comprised reflectivity measurements,

microscopy and analyses with secondary ion mass spectrometry, ion beam analysis and energy dispersive X-ray spectroscopy. The essential results are: (i) on the outer wall high reflectivity ($\sim 90\%$) is maintained for mirrors close to the channel entrance but it is degraded by $30-40\%$ deeper in the channel (ii) reflectivity loss by $70-90\%$ is measured for mirrors placed in the divertor: outer, inner and base; (iii) deuterium and carbon are the main elements detected on all mirror surfaces, the presence of beryllium is also found; (iv) thick deposits show columnar structure with some bubble-like structures, (v) the deposition in channels in the divertor cassettes is pronounced at the very entrance; (vi) photonic cleaning with laser removes deposits but the surface is damaged by laser pulses. In summary, reflectivity of all tested mirrors is degraded either by erosion with CX neutrals or by the formation of thick deposits. The implications of results obtained for first mirrors in a next-step device are discussed and critical assessment of various methods for in-situ cleaning of mirrors is presented. The conclusion is that engineering solutions should be developed in order to install shutters or to implement a cassette with mirrors to replace periodically the degraded ones.

FTP/P6-35

Detection of X-Ray from Micro-Focus Plasma (0.1 kJ)

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The X-ray emission was studied from low energy (112.5J) plasma focus device powered by a 1 mF single capacitor charged at 15 kV giving maximum discharge current of about 5 kA. The experiment was carried out on a conventional Mather-type plasma focus system and high density plasma was generated by an electrical discharge in helium gas between coaxial electrodes configuration. The X-ray emission was investigated using time-integrated and time-resolved detectors. Time-integrated X-ray pinhole images (side on) of the focus region at optimum pressure indicate a pinched volume roughly $8-10$ mm in length. Time-resolved X-ray emission profile using pin-diode with suitable absorption filter has pulse width (FWHM) ~ 100 ns.

The aim from this work is to investigate X-rays emission in a low energy Mather type plasma focus (112.5 J) using X-ray detectors.

FTP/P6-36

Mexican Design of a Tokamak Experimental Facility

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Mexico presents its Tokamak Experimental Facility design under the necessary effort to develop Science and Technology into the thermonuclear magnetic confinement fusion area. This Research and Development Project (R&D) was approved from the Mexican Education Ministry (SEP, spanish acronyms) in 2007 for its development design stage at Facultad de Ingenieria Mecánica y Eléctrica (FIME) into the Universidad Autónoma de Nuevo León (UANL). We have made this effort in order to unify and consolidate under our tokamak experimental configuration facility (in the northern Monterrey city) the Mexican Energy Fusion Program and to participate in the nearest future, in ITER development, like the Mexican Fusion Research Group.

The present design shows the development of a Tokamak Experimental Facility oriented to generate, innovate, understand, and develop scientific and technological knowledge, also to form researchers in the fusion confinement field Mexico desires with this initial effort to participate in the International Thermonuclear Experimental Reactor (ITER).

This R&D Project involves multidisciplinary physics and engineering areas, that coexists into a nuclear fusion reactor, science and technology works together to establish a natural symbiosis between theory and experiments. We decide to use hydrogen plasma in this tokamak experimental facility with the next main characteristics: major radius 41 cm (R), minor radius 18.5 cm (a), aspect ratio 2.2162 (A), safety factor 1.9552 (q), plasma current 277 kA (I_p), $\beta = 0.0532$, toroidal field 1.3 T (B_t), ionic temperature 280 eV (T_i), electronic temperature 516 eV (T_e), electronic plasma density $2 - 3 \times 10^{13}$ cm^{-}3 (n_e).

The stronger application of different sciences and technologies into our Mexican Energy Fusion Program, and the design and construction of this magnetic confinement device should allow us to develop, innovate and make significantly research exploding this new energy source concept, the magnetic confinement plasma is very complex, with this facility we can establish at first place our own schedule, a program to study and understand the different phenomena and with this aim, Mexico can participate in an integrate and intense collaboration with another nations in a great Research and Development International effort, like ITER does: A new energy source on earth.

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Innovative Concepts

ICC — Oral

ICC/1-1Ra

Turbulent Particle Pinch in Levitated Superconducting Dipole

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The Levitated Dipole Experiment (LDX) investigates the confinement and stability of plasma in a magnetic dipole field. In the experiments reported here, high-beta plasma discharges were studied when the high-field superconducting dipole magnet was levitated by attraction to a coil located above the vacuum chamber. The plasma is heated by multi-frequency ECRH. We report the observation of a plasma pinch deriving from observed turbulence. Turbulence is seen to lead to adiabatic flux tube mixing which results in profiles near to the theoretically predicted stationary density and pressure profiles. We are able to compare operations in which the internal coil was either supported or levitated. When supports are removed from the plasma, parallel losses are eliminated and cross-field transport becomes the main loss channel for both the hot and the background species. In the former configuration, losses to the supports are seen to mask the turbulence driven pinch. The density profiles are determined by use of a multi-chord interferometer and the electron temperature is estimated from X-ray measurements. Edge parameters are measured with Langmuir probes. The occurrence of stationary profiles leads to the energy confinement time exceeding the particle confinement time, a condition that is necessary for utilizing advanced fuels in a fusion power source.

ICC/P3 — Poster Session

ICC/P3-01

First Results from the Lithium Tokamak eXperiment

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The Lithium Tokamak eXperiment (LTX) is a newly commissioned, modest-scale spherical tokamak with $R = 0.4$ m, $a = 0.26$ m, and elongation of 1.5. Design targets are a toroidal field of 3.2 kG, plasma current up to 400 kA, and a discharge duration of order 100 msec. LTX is the first tokamak designed to investigate modifications to equilibrium and transport when global recycling is reduced to $10 - 20$ %. To reduce recycling, LTX is fitted with a 1 cm thick heated (300°C) copper shell, conformal to the last closed flux surface, over 85% of the plasma surface area. The plasma-facing surface of the shell will be evaporatively coated with a thin (< 100 micron) layer of molten lithium, retained by surface tension. The shell is replaceable, and a second version has been constructed, which was plasma-sprayed with $100 - 200$ microns of molybdenum to form a high-Z substrate for subsequent coating with lithium. After the installation of the second shell (in 2011), a high temperature ($500 - 600$°C) operating phase for LTX is planned. LTX is the first tokamak designed to operate with a full hot high-Z wall, near the projected operating temperature for reactor PFCs. The engineering design and construction of the hot high-Z shell, as well as the vessel and diagnostics to tolerate both lithium and 500°C internal components, will be discussed. LTX will employ short-pulse fueling with a new hydrogen molecular cluster injector, to transiently eliminate edge gas (between puffs). This fueling system will be briefly discussed. Diagnostics include single-pulse multipoint Thomson scattering, Lyman alpha arrays, microwave interferometers, spectrometers, and an edge Langmuir probe. LTX is now progressing through the shakedown phase, and first operation with a liquid lithium film wall is scheduled for Spring 2010. Later in 2010, a new diagnostic (Digital Holography) for core density variations will be tested on LTX. In 2011 a 5 A, 20 kV, 1 second hydrogen neutral beam, which will provide beam-based diagnostics, and significant core ion heating in LTX, will be installed. These upgrades will be completed in 2011, and will be described here. Finally, first results from LTX operation, with and without a full liquid lithium wall coating, will be presented.

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ICC/P4 — Poster Session

ICC/P4-01

Overview of the US Collaborative Program in Stellarator Experiments

Gates, D.A.[1], Bitter, M.[1], Boozer, A.[1], Brown, T.[1], Darrow, D.[1], Dodson, T.[1], Heitzenroeder, P.[1], Hill, K.[1], Ku, L.P.[1], Mikkelsen, D.R.[1], Neilson, H.[1], Pomphrey, N.[1], Reiman, A.[1], Skinner, C.[1], Stratton, B.[1], Zarnstorff, M.[1], Cole, M.[2], Harris, J.H.[2], Fogarty, P.J.[2], Sanchez, R.[2], Goto, M.[3], Ida, K.[3], Isobe, M.[3], Morita, S.[3], Kanaka, T.[3], Sakakibara, S.[3], Suzuki, Y.[3], Yamada, H.[3], Yoshinuma, M.[3], Baylard, C.[4], Bräuer, T.[4], Bosch, H.S.[4], Haange, R.[4], Klinger, T.[4], König, R.[4], Wegener, L.[4], Werner, A.[4], and Wolf, R.[4]

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The US collaborative program in stellarator experiments is summarized, including results The US collaborative program in stellarator experiments is summarized, including results from and contributions to both W7-X and LHD. Specific results from LHD include: turbulence calculations, fast particle physics, and high beta research. These research topics reflect areas of strength for the US program and also are areas of synergy with the priorities of the LHD research activity. Recent results from each of these areas will be presented including: lost fast-ion data during Alfvénic activity, analysis of beta limiting instabilities, and gyrokinetic stability analysis of LHD discharges. Contributions to W7-X, which is still under construction, are dominated by fusion engineering activities. These activities include: design of and assembly plans for the installation of the high temperature superconducting leads for the modular coils on W7-X, and detailed collision analysis of installations of cooling lines within the W7-X cryo-stat. These activities, along with plans for the future, will be presented in the context of the goals of the US experimental stellarator program.

ICC/P5 — Poster Session

ICC/P5-01

Active Stability Control of a High-Beta Self-Organized Compact Torus

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A magnetized coaxial plasma gun (MCPG) has been proposed as an effective device for control of a high-beta self-organized compact torus of field-reversed configuration (FRC). The initial results demonstrate that the application of an MCPG suppresses the most prominent FRC instability of the centrifugally-driven interchange mode with toroidal mode number $n = 2$. This observation was made on the Nihon University Compact Torus Experiment (NUCTE), a flexible theta-pinch-based FRC facility. In the series of experiments, a MCPG generates a spheromak-like plasmoid which can then travel axially to merge with a pre-existing FRC. Since the MCPG is mounted on-axis and generates a significant helicity, it provides the FRC-relevant version of coaxial helicity injection (CHI) that has been applied in both spheromaks and spherical tokamaks. When CHI is applied, the onset of elliptical deformation of FRC cross-section is delayed until $45-50$ μs from FRC formation compared to the onset time of 25 μs in the case without CHI. Besides delaying instability, MCPG application reduces the toroidal rotation frequency from 67 kHz to 41 kHz. Moreover, the flux decay time is extended from 57 to 67 μs. These changes have been made despite the quite modest flux content of the plasmoid: ~ 0.05 mWb of poloidal and 0.01 mWb of toroidal flux, compared with the 0.4 mWb of poloidal flux in the pre-formed FRC. The observed global stabilization and confinement improvements suggest that the MCPG can actively control the rotational instability. This global instability can also be suppressed by externally applied static multipole fields. However, it has been known that nonaxisymmetric multipole fields have adverse effects on confinement. This indicates an advantage of MCPG in that it shows both improved confinement and stability. The conventional technique does not slow the toroidal rotation down. Therefore, MCPG introduces a different stabilization mechanism that may be the same as that observed in translated FRCs [1], i.e. because of an existence of modest toroidal flux.

[1] H.Y. Guo, et al., Phys. Rev. Lett. 95, 175001 (2005).

ICC/P5-02

Two-Fluid Mechanism of Plasma Rotation in Field-Reversed Configuration

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End-shorting of the open magnetic field lines and particle loss are two mechanisms commonly considered to be sources of the ion toroidal spin-up in field-reversed configurations (FRCs). End-shorting mechanism requires conducting boundaries at the FRC ends, whereas the particle loss mechanism is related to loss of particles with preferential sign of toroidal velocity. An alternative spin-up mechanism, which does not require conducting boundaries or particle losses, is demonstrated. This mechanism relies on two-fluid description of plasma and finite parallel gradients of the plasma density. In particular, it is shown that the electron differential toroidal rotation results in generation of an antisymmetric toroidal field, which in turn leads to the ion toroidal spin-up. Time scale of the toroidal field generation is comparable to the Alfven time scale for reasonable values of the parallel density gradient. In experiments, regions of parallel density gradient can be formed, for example, near the FRC ends, where the open-filed-line plasma comes into contact with the wall, or during the FRC formation. Two-fluid simulations using the HYM code demonstrate that plasma toroidal velocity can be comparable to the ion diamagnetic velocity even in the simulations with periodic boundary conditions (ie no end-shorting) and without the particle losses.

ICC/P5-03

Recent Results from the HIT-SI Experiment

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Experiments in the Helicity Injected Torus-Steady Inductive (HIT-SI) spheromak have yielded improved current amplification and new understanding of the injector-spheromak interaction. Single injector operation shows that the two injectors have opposing, preferred spheromak current direction. A hyper-electron-viscosity model is consistent with the preferred direction, ion Doppler data, and bolometric data. The impact of this new understanding on the future direction and results from higher frequency operation are given.

ICC/P5-04

Helical-Tokamak Hybridization Concepts for Compact Configuration Exploration and MHD Stabilization

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To search for ultra-compact helical systems, a lot of exotic confinement concepts are proposed so far historically. One of the authors previously proposed the tokamak-helical hybrid called TOKASTAR in 1985 to improve the magnetic local shear near the bad curvature region. We reduced N-number (toroidal mode number) and proposed $N = 1$ and 2 TOKASTAR configurations. This is characterized by simple and compact coil systems with enough divertor space relevant to reactor designs. Based on this TOKASTAR concept, an $N = 2$ C-TOKASTAR (Compact Tokamak-Helical Hybrid) machine was constructed. The rotational transform of this compact helical configuration is rather small, but can be utilized as a compact electron plasma machine for multi-purposes. The C-TOKASTAR ($R \sim 3.5$ cm) has double spherical winding helical coils and a pair of poloidal coils. Existence of magnetic surface and electron confinement property in C-TOKASTAR device were investigated by an electron-emission impedance method. In this method, an electron gun filament is inserted into C-TOKASTAR and the electron current is detected. When the poloidal coil current was scanned under the steady helical coil current, a clear decrease in the emission current signal was observed at the onset of impose of the poloidal field. It suggests the increase of circuit impedance because the magnetic surface is formed and the electrons are trapped. Calculation of the particle orbit also supports that closed magnetic surface is formed in the cases that the ratio between poloidal and helical coil current is appropriate. Another aspect of the research using TOKASTAR configuration includes the evaluation of the effect of the outboard helical field application to tokamak plasmas. It is considered that outboard helical field has roles to assist the initiation of plasma current, to improve MHD stability, and so on. To check these roles, we made TOKASTAR-2 machine ($R \sim 12$ cm, $B \sim 1$ kG) with ohmic heating central coil, eight toroidal field coils, a pair of vertical field coils and two outboard helical field coil segments. The ECH (2 kW) plasma start-up and plasma current disruption control experiments might be expected in this machine. According to the preliminary experiment, we confirmed the effect of outer helical field application to modify bursting density fluctuation to repetitive mild oscillations.

ICC/P5-05

Modeling of High Density and Strong Magnetic Field Generation by Plasma Jet Compression

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The main topic is the generation of dense magnetized plasmas and ultra-strong magnetic field for fusion. The principal theoretical result is a demonstration of the feasibility of electromagnetic-driven spherical liner implosions in the cm/μs regime. The field reversed configuration (FRC) and the cusp geometry of the magnetic field (antiprobkotron) are alternate systems with attractive prospects. Preferable choice of plasma confinement in magnetic device may be combined with properties of inertial confinement. The main aim of this work is physical analysis of high temperature plasma (target) confined by magnetic field with taking into account plasma guns (liner) to push and compress plasma of compact configuration. This study points to solution of problem connected with theoretical investigation of methods supplying with magneto inertial fusion (MIF). We consider the value corresponding to the equality of the plasma pressure to the magnetic pressure as the effective magnetic field. Numerical calculation of temporal evolution of the particle balance and plasma power balance were performed to define the minimal requirements for magnetic field on the final stage of the compression. The results show for most prospective regimes minimum effective values $B \sim 300$ T are needed. Results of numerical simulation are presented.

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ICC/P5-06

The upgrade to the Mega Amp Spherical Tokamak

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MAST has made valuable contributions to Spherical Tokamak (ST) development and general tokamak physics. A new upgrade has been designed and a first major stage is being implemented, allowing substantial progress on critical issues for a Component Test Facility (ST-CTF) and contributions to ITER and DEMO physics. The first stage upgrade machine will be operational in late 2015 with two main goals: test of a novel, reduced heat load, Super-X Divertor (SXD) for DEMO and ST studies; and dominantly non-inductive plasmas with current $>$ 1 MA for two to three current diffusion times. MAST's divertors, currently open with no density control, will be rebuilt as closed and cryogenically pumped. New coils will control the outer legs in either SXD or classic X-point configurations. The SXD will aid detached regime access by greatly increased connection length, in a scrape-off-layer region separated from the main plasma, and by stretching the outer leg to 1.5 m radius (cf. X-point radius 0.8 m). Initial modelling shows outer target temperature reduced by a factor $\sim$ 3 cf. the conventional divertor. Gas puffing will further cool the divertor aiding detachment.

Non-inductive scenarios will be accessed with, in the first stage, three 2.5 MW NB sources (two located for off-axis NBCD). Simulations show fully non-inductive scenarios with $I_p > 1$ MA should be accessible, as well as relaxed q-profiles with minimum $q > 2$ (stability analyses will be reported). A new centre column with 70% more flux and a long-pulse TF field power supply will allow 50% higher field (0.9 T at $R = 0.7$ m) and several second pulse lengths at current $>$ 1 MA. The stage one upgrade will widen operational space to lower collisionality, higher fast particle beta, current up to 2 MA, and stronger shaping (elongation to 2.5, triangularity to 0.6). A second upgrade stage addressing other key ST-CTF physics issues will be described, including NB H&CD upgraded to 12.5 MW, EBW, pellet injector fuelling, and external ELM coils.

ICC/P7 — Poster Session

ICC/P7-01

Kinetic Behaviors of Energetic Ions in Oblate Field-Reversed Configuration

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Energetic ions are keys to improve stability and confinement of the field-reversed configuration (FRC) plasma. In this paper, kinetic behaviors of energetic ions in FRC plasmas are comprehensively investigated using spontaneously driven sheared toroidal flow and tangential neutral beam injection (NBI). An oblate FRC is produced by merging of two spheromak plasmas with opposing toroidal magnetic field (counter-helicity merging) in neon, argon and krypton discharges in TS-4 device. During the counter-helicity merging, significant ion heating is provided by magnetic reconnection of both poloidal and toroidal fields. The reconnection outflow also has both radial and toroidal components, resulting in generation of sheared toroidal flow whose direction is determined by the polarity of the merging two spheromaks. Since the accelerated ion's kinetic energy reaches $3-5$ times larger than the initial ion temperature, the ion's gyroradius becomes comparable to the plasma size. The experimental results show that the decay rate and the electron density profile of the oblate FRC are strongly affected by the direction of the toroidal flow. When the direction of the toroidal flow is parallel to the plasma current on the outside of the magnetic axis, the electron density peaks at the inboard side and the FRC shows small energy decay rate. However, when the toroidal flow is anti-parallel to the plasma current on the outside of the magnetic axis, the electron density peaks near the plasma edge with large gradient and the FRC suffers from rapid decay especially for heavy ions. Numerical calculation of the fast ion's trajectory shows that most of the ions in the latter case drift toward outboard side and escape from the plasma region in a short period, indicating that the differences in the decay rate and the density profile are mainly induced by the preferential particle loss of the energetic ions. Confinement of the energetic ions is also essential for the NBI heating of the FRC plasma. Hydrogen beam with energy of $\sim$13 keV and current of $\sim$5 A was injected tangentially to the oblate FRC plasma for the first time. The observed improvement of the magnetic energy decay rate is much larger than the input NBI power, suggesting that global stability of the FRC is improved by the NBI fast ions.

ICC/P7-02

Improvement of Cusp Type and Traveling Wave Type Plasma Direct Energy Converters Applicable to Advanced Fusion Reactor

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An advanced fusion reactor has a great advantage in the view point of social receptivity on radioactive waste disposal. As a commercial plant, an application of direct energy converter (DEC) is essential for economical payment. A system consisting of cusp (CUSP) DECs and traveling wave (TW) DECs in a FRC-based D-^{3}He reactor was conceptually designed previously, and conversion efficiencies of 65% of CUSPDEC and 76% of TWDEC provide 1GW electricity. In the scenario of the DEC system development, the proof-of-principle experiments by small-scale simulators had been done by the authors, and the extension of working range of plasma parameters is the current issue. Three subjects are treated in the paper: A) charge separation of high density influx and B) multi-stage deceleration for CUSPDEC, and C) energy broadening of influx for TWDEC. About A), a slanted cusp magnetic field proposed by authors can discriminate more efficiently than a normal cusp field, which was experimentally confirmed previously for relatively low density plasma (10^{12} m^{-3}). In higher density plasma ($\sim 10^{16}$ m^{-3} in the commercial device), a collective behavior may obstruct particle separation. The separation efficiency, and thus the conversion efficiency degrades with the ratio of plasma radius to Debye length. The efficient working of the slanted cusp field with an additional measure has been confirmed up to 10^{15} m^{-3}. Based on the idea of B), energy of slow ions reflected by high potential due to electrode bias and/or piled high density ions is recovered by an additional lateral collector. Following to the previous rough estimation of efficiency, the paper presents a precise one taking the radial position and energy of incident ions into account. The scheme provides a better design of the additional collector, by which 10% efficiency increment is obtained. As for C), "fan-type TWDEC" was proposed previously, which recovered degradation of efficiency due to energy broadening of influx. The paper shows an experimental verification of the concept. A new simulator "dual-beam TWDEC" has been constructed, which has the same concept with fan-type TWDEC and reduced structure: two different energy beams are used instead of an influx with continuously broadened energy. Dual beams are successfully extracted and following works are in progress.

IFE

Inertial Fusion Experiments and Theory

IFE — Oral

IFE/1-1

High-Performance ICF Target Implosions on OMEGA

Meyerhofer, D.D.[1], McCrory, R.L.[1], Betti, R.[1], Boehly, T.R.[1], Casey, D.T.[1], Collins, T.J.B.[1], Craxton, R.S.[1], Delettrez, J.A.[1], Edgell, D.H.[1], Epstein, R.[1], Fletcher, K.A.[1], Frenje, J.A.[1], Glebov, Y.Y.[1], Goncharov, V.N.[1], Harding, D.R.[1], Hu, S.X.[1], Igumenshchev, I.V.[1], Knauer, J.P.[1], Li, C.K.[1], Marozas, J.A.[1], Marshall, F.J.[1], McKenty, P.W.[1], Nilson, P.M.[1], Padalino, S.P.[1], Petrasso, R.D.[1], Radha, P.B.[1], Regan, S.P.[1], Sangster, T.C.[1], Séguin, F.H.[1], Seka, W.[1], Short, R.W.[1], Shvarts, D.[1], Skupsky, S.[1], Soures, J.M.[1], Stoeckl, C.[1], Theobald, W.[1], Yaakobi, B.[1], and Collaborators from MIT and SUNY-Geneseo

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The Omega Laser Facility is used to study inertial confinement fusion (ICF) concepts. This talk describes progress in direct-drive central hot-spot (CHS) ICF, shock ignition (SI), and fast ignition (FI). Omega is the only facility worldwide performing cryogenic target implosions required for most ICF target designs. Cryogenic deuterium-tritium (DT) target implosions on OMEGA have demonstrated the highest DT areal densities measured in ICF implosions. The Lawson's performance parameter P_τ (pressure times energy confinement time) is inferred from ion temperature, areal density, and neutron-yield measurements. For implosions with ion temperatures of ~ 2 keV, areal densities of $0.2-0.3$ g/cm^2, and neutron yields of $5-6\times10^{12}$, the Lawson's parameter is $P_\tau \sim 1$ atm-s, comparable to that obtained in DT discharges on the Joint European Tokamak.

A multiple-picket direct-drive-ignition design that generates four shock waves (similar to the NIF baseline indirect-drive design) is being tested on OMEGA. In 1D it produces a gain of 48 for a 1.5-MJ implosion. The shock-wave timing was optimized using cryogenic cone-in-shell shock-timing targets. Cryogenic-DT-target implosions have produced record areal densities of ~ 0.3 g/cm^2 that are 50% higher than those reported at the 2008 IAEA FEC conference and test the diagnostics required for the ignition campaign on the NIF. Further performance improvements are expected this summer as a result of beam smoothing upgrades currently being installed.

SI and FI are two-step concepts where the fuel is assembled by one laser pulse followed by a second pulse that initiates ignition, allowing the possibility of higher gains than CHS ignition. SI uses an intensity spike at the end of the same laser pulse that compresses the target, while FI requires a high-intensity, high-energy petawatt beam (HEPW) to provide the electron or ion beam needed to ignite the compressed fuel. The Omega EP Laser Facility provides kilijoule-class HEPW pulses for integrated FI experiments, which have doubled the neutron yield in a cone-in-shell FI deuterated plastic-shell implosion. Initial SI experiments have shown significantly increased neutron yields and areal densities compared to those without the SI high-intensity spike. Various FI and SI configurations will be further tested on OMEGA during the summer of 2010.

IFE/1-2

Integrated Experiments of Fast Ignition with Gekko-XII and LFEX Lasers

Shiraga, H.[1], Fujioka, S.[1], Koga, M.[1], Nakamura, H.[1], Watari, T.[1], Shigemori, K.[1], Sakawa, Y.[1], Nishimura, H.[1], Tanaka, K.A.[2], Habara, H.[2], Ozaki, T.[3], Isobe, M.[3], Homma, H.[1], Fujimoto, Y.[1], Nakai, M.[1], Norimatsu, T.[1], Iwamoto, A.[3], Mito, T.[3], Miyanaga, N.[1], Kawanaka, J.[1], Jitsuno, T.[1], Tsubakimoto, K.[1], Nakata, Y.[1], Sarukura, N.[1], Nagatomo, H.[1], Johzaki, T.[1], Murakami, M.M.[1], Mima, K.[1], Sakagami, H.[3], Sunahara, A.[4], Taguchi, T.[5], Nakao, Y.[6], and Azechi, H.[1]

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Based on the successful result of fast heating of a shell target with a cone for heating beam injection, FIREX-1 project has been started from 2004. Its goal is to demonstrate fuel heating up to 5 keV by using upgraded heating laser beam. For this purpose, LFEX laser, which can deliver, at the full spec, an energy up to 10 kJ in a $0.5 - 20$ ps pulse, has been constructed beside Gekko-XII laser system at the Institute of Laser Engineering, Osaka University. It has been activated and became operational since 2009. Instead of the previous experiment with PW laser, upgraded integrated experiments of Fast Ignition have been started by using LFEX laser with an energy up to 1 kJ in a $1-5$ ps 1.053-micron pulse. Initial experimental results including implosion of the shell target by Gekko-XII, heating of the imploded fuel core by LFEX laser injection, and up to 30 times increase of the neutron yield due to fast heating compared with no heating have been achieved. Results indicate that the heating efficiency is $3 - 5\%$, which is much lower than $20 - 30\%$ expected from the previous data. It is attributed to the too hot electrons generated in a long scalelength plasma in the cone preformed with a prepulse in the LFEX beam. By controlling the prepulse condition, 5-keV heating can be expected at full output of LFEX laser with improved heating efficiency.

IFE/1-3

The Path to Inertial Fusion Energy

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The 192-beam National Ignition Facility (NIF) at the Lawrence Livermore National Laboratory (LLNL) in Livermore, CA, is operational. Results from initial NIF ignition program experiments have been very promising. In light of this strong progress and the impending start of layered target ignition experiments at NIF later this year, the U.S., EU, Japan, Russian Federation, P.R. China and other nations are currently examining the implications of NIF ignition for inertial fusion energy (IFE). The FIREX and HiPER projects in Japan and the EU, respectively, are looking into fast ignition and other fusion alternatives as well as efforts in laser, optical, target fabrication, target chamber, and other technology development required for IFE. A laser-based IFE power plant will require a repetition rate of 20 Hz and approximately 10% electrical efficiency, compared to the 10^{-4} Hz and 0.5% efficiency characteristic of NIF. The realization of IFE will also require further development and advances of large-scale target fabrication, target injection, target chamber, and other technologies.

While progress towards ignition has received significant attention, it is important to realize that many of the scientific and technological advances made under the NIF ignition program in areas such as laser technology, target fabrication, and diagnostics also directly support the advancement of IFE. Examples include the demonstration of multi-pass laser architecture, the development of robust capsule and hohlraum manufacturing techniques and target layering capability, and the development of hardened diagnostics suitable for the extreme ignition environment. These results coupled with a robust IFE program could lead to a prototype IFE demonstration plant in the $10-15$ year time frame. LLNL, in partnership with other institutions, is continuing to examine the Laser Inertial Fusion Engine (LIFE) concept and is looking in detail at issues such as the use of second vs. third harmonic laser light and the advantages of both pure fusion and fusion-fission schemes.

This talk will review these topics as well as the overall progress of IFE technology worldwide and describe a $10-15$ year plan for IFE generally. It is important that the IFE community develop an effective and implementable plan in the near term to be ready to immediately capitalize on the demonstration of ignition at NIF.

IFE/1-4

Lasers for Inertial Fusion Energy

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The achievement of controlled fusion burn and gain with mega-joule-class laser facilities such as the National Ignition Facility lays the foundation for future energy systems based on inertial confinement fusion. The technical challenges for laser inertial fusion energy systems are numerous and include the generation and operation of cost-effective, robust, mega-joule lasers that can operate at repetition rates that are more than 100 000 times beyond that of today's fusion demonstration laser systems. In this presentation we will review the 20+ year development of high power laser technologies for inertial fusion energy and the new laser architectures based on these technologies that will enable fusion energy generation via inertial confinement fusion.

This work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344.

IFE/1-5

Chamber Dynamics and Radiological Protection in HiPER IFE Project under Repetitive Operation

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HiPER, High Power laser Energy Research, the proposed European Laser-Driven Fusion Energy Facility will finish its Preparatory Phase as ESFRI European programme next year. Key decisions to be taken in this phase concern the type of facility and the future use of it. As next step, the involved European Teams will start working on a strategy to collect and improve the knowledge on Energy generation by Inertial Fusion. A key decision has been the definition of a facility with a repetitive laser operation with Fast Ignition as a primary option in a preliminary realistic burst mode of hundred to thousand shots at $5-10$ Hz rate. The Engineering Test Bed results obtained with this approach will be able to demonstrate, with the lowest risk, repetitive laser-injection systems in a new model of Chamber. These new conditions assume the ability to accommodate Experiments in Technology relevant for HiPER as a power production facility. This paper shows the differences in designing the chamber when going from single shot (NIF or LMJ Ignition/Gain machines) to repetitive systems not only in Laser requirements but also in Chamber area, activation, damage in optics, wall and structural materials. The HiPER project will design an Inertial Fusion Facility for ENERGY generation based on Fast Ignition (shock ignition is under consideration) and repetition. Demonstration of Energy production (ignition and gain) will be achieved with a minimum of 48 compression beams giving 400 kJ in tens of nanoseconds and a PW beam line of 50 kJ in tens of picoseconds. NIF ignition (and LMJ after), early realization of DPSSL repetitive lasers, and progress in Repetitive Injection of target and manufacturing are basic for making this proposal attractive on time. Different activities in basic physics, astrophysics, nuclear physics, high density matter are expected. We present a study on the Chamber area which considers aspects such as: i) resistance to debris, radiation, shrapnel and neutrons effects; ii) deep vacuum and cryo environments required for experiments; iii) accommodation of the many diagnostic instruments, beam-lines, and associated optics and equipment; iv) minimization of the activation of the material components of the chamber together with low radioactivity and ease of operation, maintenance and chamber disposal; v) tritium handling.

IFE/P6 — Poster Session

IFE/P6-01

Core Heating Scaling for Fast Ignition Experiment FIREX-I

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In Institute of Laser Engineering (ILE), Osaka University, a 4-beam bundled new ultra-intense high-energy laser LFEX (Laser for Fast-ignition Experiment) has been constructed, and FIREX-I (Fast-Ignition Realization Experiment Project, Phase-I) has been started. The final goal of FIREX-I is demonstration of core heating up to 5 keV using 10 kJ heating laser. Previously, we have carried out the integrated simulations, which reproduced the core heating properties of the PW experiments (2002) and showed the importance of pre-plasma in core heating performance. In the present paper, on the basis of the integrated simulations, we analyze the first integrated experiments using LFEX laser and evaluate the requirements for achieving the final goal of FIREX-I.

IFE/P6-02

Hot Electron Generation for Fast Ignition

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Fast ignition (FI) inertial confinement fusion, by separating the compression and ignition steps, offers high gain compared to the central hot spot concept and puts relatively less stringent requirement on the symmetric compression of the fuel. For electron ignition the efficient conversion of laser energy into fast electrons and transport to the compressed fuel is of importance. The Titan laser facility at LLNL has been used to investigate critical aspects of this process: photon conversion to useful electron energies, electron divergence in a cone structure in the presence of pre-existing plasma, and propagation into hot plasma. New diagnostics and target types were developed to allow investigation under FI relevant conditions. We find that the ignition-laser-prepulse-induced modification of the laser plasma interface has a strong influence on the parameters of laser-produced hot electrons. Even small amounts (of order 10 mJ) cause the electron spectrum to be hotter and more divergent than otherwise. It has also been observed that geometry of a narrow cone tip increases these effects. Hybrid PIC modeling has been performed to understand the underlying physics that plays an important role in these processes.

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IFE/P6-03

Experimental Demonstration of Key Parameters for Ion-based Fast Ignition

Hegelich, B.M.[1], Albright, B.J.[1], Fernandez, J.C.[1], Gautier, D.C.[1], Huang, C.K.[1], Jung, D.[1], Kwan, T.J.[1], Letzring, S.[1], Palaniyappan, S.[1], Shah, R.C.[1], Wu, H.C.[1], Yin, L.[1], Honrubia, J.J.[2], Temporal, M.[2], Steinke, S.[3], Schnürer, M.[3], Sokollik, T.[3], Sandner, W.[3], Henig, A.[4], Hörlein, R.[4], Kiefer, D.[4], and Habs, D.[4]

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Research on fusion fast ignition (FI) initiated by laser-driven ion beams has made substantial progress in the last year. Compared to electrons, FI based on a beam of quasi-monoenergetic ions (protons or heavier ions) has the advantage of a more localized energy deposition, and stiffer particle transport, bringing the required total beam energy close to the theoretical minimum of $\sim$ 10 kJ. High-current, laser-driven ion beams [1] are excellent for this purpose, and because of their ultra-low transverse emittance, these beams can be focused to the required dimension, $\sim$ 10 s mu. Because they are created in ps timescales, these beams can deliver the power required to ignite the compressed D-T fuel, $\sim$ 10 kJ / 50 ps. Our recent integrated calculations of ion-based FI include high fusion gain targets and a proof of principle experiment. The simulations identify three key requirements for the success of Ion-driven Fast Ignition (IFI): 1) the generation of a sufficiently high-energetic ion beam ($400 - 500$ MeV for C), with 2) less then 20% energy spread at 3) more than 10% conversion efficiency of laser to beam energy. This paper describes new results from experiments at Los Alamos and Berlin, demonstrating all three parameters in separate experiments. Using diamond nanotargets and ultrahigh contrast laser pulses we were able to demonstrate $>$ 500 MeV carbon ions at the Trident laser at Los Alamos [2], as well as carbon pulses with $<$ 20% spread at $40 - 100$ MeV. At the Max-Born-Institute's 50 TW Ti:Sapphire laser we were able to demonstrate 80 MeV carbon ions at $>$ 10% conversion efficiency accelerated by laser pulses with as little as 0.7 J energy [3]. Furthermore we are presenting numerical and theoretical results towards the integration of all three parameters and first results of new laser developments regarding contrast and pulse shaping necessary to fulfill all three requirements at once. This paper summarizes the recent progress in ion-based FI research and outlines future steps towards a final feasibility assessment of this concept.

[1] B.M. Hegelich, et al., Nature 439, 441 (2006).

[2] B.M. Hegelich, et al., in prep. (2010)

[3] Henig et al., PRL 103, 245003 (2009)

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IFE/P6-04

Progresses of Impact Ignition

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In impact ignition scheme, a portion of the fuel (the impactor) is accelerated to a super-high velocity, compressed by convergence, and collided with a precompressed main fuel. This collision generates shock waves in both the impactor and the main fuel. Since the density of the impactor is generally much lower than that of the main fuel, the pressure balance ensures that the shock-heated temperature of the impactor is significantly higher than that of the main fuel. Hence, the impactor can reach ignition temperature and thus become an igniter.

Here we report major new results on recent impact ignition research: (1) A maximum velocity ~ 1000 km/s has been achieved under the operation of NIKE KrF laser at Naval Research Laboratory (laser wavelength = 0.25 μm) in the use of a planar target made of plastic and (2) We have performed two-dimensional simulation for burn and ignition to show the feasibility of the impact ignition.

It is concluded that, by increasing the impactor velocity from 700 km/s to over 1500 km/s, and the energy efficiency from 2% to 10%, we estimate the laser energy necessary to create a hot spot to ignite DT fusion fuel to be $200 - 300$ kJ, which is much lower than that required for central ignition. Scaling up of the present experiments will demonstrate that full-scale impact ignition could be attained within a practical range.

IFE/P6-05

Direct-Drive Concept for Z-Pinch Inertial Fusion

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X-ray-driven central hot-spot ignition of a spherical fusion capsule is the baseline approach for Inertial Confinement Fusion (ICF) being pursued in the United States. While research using x-rays produced by Z-pinches to drive these spherical fusion capsules has demonstrated radiation pulse shaping and the required uniformity of the radiation flux incident on the fusion capsule, the 3 MJ of x-ray energy produced on the Z facility is not large enough to drive a spherical capsule to ignition and high yield. An alternate approach to pulsed-power ICF is Magnetized Liner Inertial Fusion, where the high current directly compresses a solid cylindrical liner containing preheated, magnetized DT fuel to ignition and fusion burn conditions. Our simulations indicate that the efficiency of coupling the driver energy to the fusion fuel in these direct magnetically-driven approaches is more than an order of magnitude greater than x-ray-drive concepts. This increased efficiency could greatly reduce the pulsed-power driver requirements for high yield fusion. Numerical simulations indicate that significant fusion yields may be obtained on the Z facility through 25-MA-driven, 100-ns implosions of cylindrical metal liners if the DT fuel is magnetized to greater than 10 Tesla and preheated by the Z-Beamlet laser to $100 - 500$ eV. Since the Magneto-Rayleigh-Taylor (MRT) instability poses a significant threat to this approach to fusion, we have recently performed initial Z experiments to test the robustness of these liner implosions to the MRT instability and validate our multi-dimensional calculations of the Z-pinch implosion. To provide increased efficiencies in the pulsed power driver itself, we are assembling a small module of multiple Linear Transformer Driver (LTD) cavities to study system scaling and reliability. This repetitive LTD technology may be more than twice as efficient as previous technologies and be a key to developing pulsed-power-driven inertial fusion energy.

IFE/P6-06

Progress in U.S. Heavy Ion Fusion Research

Logan, B.G.[1], Barnard, J.J.[1], Bieniosek, F.M.[1], Cohen, R.H.[1], Davidson, R.C.[1], Efthimion, P.C.[1], Friedman, A.[1], Gilson, E.P.[1], Grisham, L.R.[1], Grote, D.P.[1], Henestroza, E.[1], Kaganovich, I.D.[1], Kwan, J.W.[1], Lee, E.P.[1], Leitner, M.A.[1], Lidia, S.M.[1], Lund, S.M.[1], More, R.M.[1], Ni, P.[1], Perkins, L.J.[1], Qin, H.[1], Roy, P.K.[1], Seidl, P.A.[1], Sharp, W.M.[1], Startsev, E.A.[1], and Waldron, W.L.[1]

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New and unpublished results will be reported on ion-beam-driven warm dense matter (WDM target physics on the Neutralized Drift Compression Experiment (NDCX-I) using polarization-sensitive fast optical pyrometry, on intense beam neutralization in strong magnetic fields, on simulations with novel pulse compression to a few-hundred picoseconds in a new, 100 times more powerful facility (NDCX-II) now under construction at Lawrence Berkeley National Laboratory, and on first 1D implosion calculations of shock ignition using heavy ion beams. These advances define international research opportunities using the 100 times larger beam energies in NDCX-II when completed in 2012.

IFE/P6-07

Advantages of KrF Lasers for Inertial Confinement Fusion Energy

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Advanced concepts for direct drive inertial confinement fusion (ICF) have emerged that may lead to significant gain ($g > 100$) while reducing the total energy requirements of the laser ($E \sim$500 kJ). For example, recent shock ignition designs use high intensity laser pulses (10^{15} W/cm^2) to compress low aspect ratio pellets then apply a final high intensity spike (10^{16} W/cm^2) to ignite the fuel. These analyses were based on an excimer laser with a krypton-fluoride lasing (KrF) medium. KrF systems are particularly well suited to these new ideas as they operate in the ultraviolet (248 nm), provide highly uniform illumination, possess large bandwidth ($1-3$ THz), and can easily exploit beam zooming to improve laser-target coupling for the final spike pulse. While this driver option is strongly supported by a multitude of attractive technological features, decisive factors for any candidate technology must assess the resulting target physics as necessary for high gain operation. This presentation will examine advantages of KrF lasers in relation to the new implosion designs. Supporting experimental and theoretical studies of hydrodynamic instabilities and laser plasma interactions (LPI) conducted by the Nike laser group at the U. S. Naval Research Laboratory will be discussed. Recent studies of the two plasmon decay instability have made the first successful determination of the threshold intensity for this instability in an ICF relevant plasma driven by a KrF laser. This instability is a major concern due to its ability to generate hot electrons that may preheat the compressed fuel prior to ignition. The observed threshold intensities are higher than theoretical estimates (neglecting collisional and Landau damping effects) and higher than reported for longer wavelength lasers. An expanded range of allowed intensities would help KrF lasers meet the requirements of advanced ICF designs. Recent experimental work has also shown that the high ablation pressures and smooth profiles obtained with the Nike laser can be used to accelerate planar targets to velocities consistent with the requirements of impact ignition.

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IFE/P6-08

Measurements of Spontaneous Electromagnetic Fields, Plasma Flows, and Implosion Dynamics in Indirect-Drive Inertial-Confinement Fusion

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Measurements of x-ray-driven implosions with charged particles have resulted in the quantitative characterization of critical aspects of indirect-drive inertial fusion. Three types of spontaneous electric fields differing in strength by two orders of magnitude, the largest being nearly one-tenth of the Bohr field, were discovered with time-gated proton radiographic imaging and spectrally-resolved proton self-emission. The views of the spatial structure and temporal evolution of both the laser drive in a hohlraum and implosion properties provide essential insight into, and modeling validation of, x-ray-driven implosions.

IFE/P6-09

Proton Radiography and Fast Electron Propagation through Cylindrically Compressed Targets

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This experiment was achieved on the VULCAN laser facility (UK) using four long pulse beams ($\sim 4 \times 70$ J, 1 ns, at 0.53 μm) to compress a hollow plastic cylinder filled with plastic foam of three different densities (0.1, 0.3 and 1 g/cc). Protons accelerated by a ps laser pulse have been used to radiograph the imploding cylinder (degree of compression and stagnation time). Results were compared to those from hard X-ray radiography. Finally, Monte Carlo simulations of proton propagation in the cold and in the compressed targets allowed a detailed comparison with 2D numerical hydro simulations. In the second phase of the experiment a short pulse (10 ps, 160 J) beam generated fast electrons that propagate through the compressed matter by irradiating a nickel foil at an intensity of 5×10^{18} W/cm^2. X-ray spectrometer and imagers were implemented in order to estimate the compressed plasma conditions and to infer the hot electron characteristics. Results are discussed and compared with simulations.

IFE/P6-10

Control of Instabilities due to some Transient Processes in Laser Accelerated Target

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Various types of instabilities appear at the laser induced ablation surface due to laser non-uniformity, target roughness, RT, RM, KM etc. and hamper fusion reaction efficiency. Therefore controlling instabilities is one of the important issues in direct drive related to IFE. Interestingly, small scale non-uniformities less than $2-3\,\mu$m for large scale facility like PALS, Prague can be smoothened and not large scale wavelengths. Such long wavelength non-uniformities appearing on the ablation surface later imprint on the compressed DT fuel core. In our on-going research work, we propose to use low density plastic targets embedded with micro particle ($1-3\,\mu$m in dia.) to break initial non-uniformities appearing at the ablation surface. Our experimental results using plastic targets seeded with aluminium, gold and tungsten micro-particles show uniform motion of the acceleration of target which was recorded using optical shadowgraphy. Such target structure can break long scale wavelength instabilities in to smaller scale at the ablation surface and smoothen by mitigating their growth. There could be several transient processes in seeded targets acting simultaneously and supporting the instability smoothing including micro-particles acting as mechanical obstacle, vortex formation, self-generated magnetic field due to vortices etc. In the preliminary analysis we assume that the inter-particle spacing d setup by the micro-particles restrict the easy flow of the instabilities, thereby limiting their wavelength. We generate all possible sinusoidal waves and the wavelength of perturbation is adjusted according to the modes within the fluid boundary. Resultant R of all these modes provide the amplitude of the instability which is responsible for the breakup of the accelerated target. We calculate R for various values of d. Results show that the instability amplitude can be suppressed to $3\,\mu$m when d is varied up to $10\,\mu$m. By using a known thickness of seeded target it will be further possible to break the instability wavelength and reduce the instability amplitude to the required value. Inter-particle spacing d can be controlled by using needed amount of the micro-particles n. We are studying the above mentioned transient processes by simulation. This work is important to direct drive implosion. Details of the work will be presented.

IFE/P6-11

Measurements of the Deuterium-Tritium Branching Ratio using ICF Implosions

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Fusion of deuterium (D) and tritium (T) produces an excited $^5He^*$ nucleus, which has several possible de-excitation modes. The most common mode results in the emission of a 14.1 MeV neutron and a 3.5 MeV alpha particle. A much less frequent mode involves the excited $^5He^*$ nucleus relaxing to the ground state via the emission of a 16.75 MeV gamma ray. Historically, interest has been focused on the measurement of DT branching ratio. All existing measurements have been accelerator-based (> 100 keV) and compromised by target design difficulties and intractable 14.1 MeV neutron-induced radiation backgrounds.

Recent advances in Cherenkov diagnostics and Monte Carlo simulations developed at Los Alamos National Laboratory (LANL) have allowed for more precise DT branching ratio measurements. The high-bandwidth of the Gas Cherenkov Detector (GCD) allows for the detection of DT fusion gamma rays before the 14.1 MeV neutrons arrive at the GCD, relieving the measurement of undesired neutron background. We report a preliminary measurement of the DT branching ratio of $(2.6 \pm 1.1)10^{-5}$ as measured at the OMEGA laser facility at the University of Rochester, NY. To infer the DT branching ratio, various target materials such as Si, SiO_2, Al, Al_2O_3, or Cu are placed near target chamber center (TCC). 14.1 MeV neutrons interact with the targets, otherwise called pucks, and produce secondary, neutron-induced gamma-rays, which are used to calibrate the GCD. In conjunction with an independent neutron yield measurement and a coupled MCNP/ACCEPT simulation code, the detection efficiency of the GCD has been estimated. Finally, the absolute DT fusion gamma-ray signal has been determined and resultant DT branching ratio is inferred. Improved measurements of the DT branching ratio are beneficial to gamma-ray-based DT burn-history measurements for the National Ignition Facility (NIF) and other facilities contemplating the need for similar diagnostic information.

IFE/P6-12

Implosion Physics and Robust Target Design for Fast Ignition Realization Experiment

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Fast ignition is an attractive scheme in laser fusion. At the first phase of the scheme, highly compressed fuel core plasma is formed by implosion laser, and then, just around the maximum density time, it is heated by heating laser to achieve a fusion burning condition. A shell targets for Fast ignition is fitted with reentrant gold cone target to

preserve a path for heating laser. Therefore, understanding of the cone guided implosion dynamics is one of significant problem. We have been studying these physics and design, especially a formation of high density core plasma, and breakdown of the tip by using two-dimensional radiation hydrodynamics simulation. In these work, we have found that RTI causes tip breakdown in early phase, and decrease of fuel core plasma density. A dynamics of massive jet which hit the tip is evaluated also. These serious problems can be avoided by using the sophisticated laser pulse shape and irradiation pattern for slow velocity, and low adiabat implosion. In this paper, we will describe design principles of Fast ignition which are revalidated with the simulations, and more robust target design is proposed for Fast ignition experiment.

IFE/P6-13

Investigation of High Gain, Shock-Ignited Targets on the National Ignition Facility for Near Term Application

Perkins, L.J.[1], Betti, R.[2], Schurtz, G.[3], Craxton, R.S.[2], Dunne, A.M.[4], LaFortune, K.[1], Schmitt, A.[5], McKenty, P.[2], Bailey, D.[1], Lambert, M.[1], Ribeyre, X.[3], Atzeni, S.[9], Theobald, W.[2], Strozzi, D.[1], Andersen, K.[2], Casner, A.[6], Murakami, M.M.[7], and Comley, A.[8]

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Shock ignition, a new concept for igniting thermonuclear fuel, offers the potential for a near-term test of high gain inertial confinement fusion on the National Ignition Facility at less than 1MJ drive energy and without the need for new laser hardware. In shock ignition, compressed fusion fuel is separately ignited by a strong spherically-converging shock and, because capsule implosion velocities are significantly lower than those required for conventional hotpot ignition, simulations indicate that high fusion energy gains of 60 may be achievable on NIF at laser drive energies around 0.5 MJ, extending to 100 at 1 MJ. Because of the simple all-DT target design, its in-flight robustness, the potential need for only 1D SSD beam smoothing, minimal early-time LPI preheat issues, and employment of day-1 laser hardware, this target may be easier to field on NIF than a conventional (polar) direct drive hotspot ignition target. Like fast ignition, shock ignition has the potential for high fusion yields at low drive energy, but requires only a single laser with less demanding timing and spatial focusing requirements. Of course, conventional symmetry and hydrodynamic stability constraints will apply. In general, shock ignition offers the prospects for high-gain targets that may lead to smaller, more economic fusion power reactors and a cheaper fusion energy development path. In this paper we will present target performance simulations, delineate the critical issues and describe the immediate-term R&D program that must be performed in order to test the potential of a high gain shock ignition target on NIF in the near term.

IFE/P6-14

Ignition Regime and Burn Dynamics of DT-Seeded D^3He Fuel for Fast Ignition Inertial Confinement Fusion

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The feasibility of igniting D^3He fuel in the fast-ignition inertial confinement fusion is examined by appropriately treating the plasma heating processes by fusion-produced fast particles. Simulations have been made for a DT/D-^{3}He pellet compressed to $2000 - 5000$ times the liquid density. Newly included is the transport of recoil ions generated by knock-on collisions of D-T neutrons and D-^{3}He protons. Although the ignition condition is degraded by inclusion of the recoil-ion transport, it is still possible to obtain sufficient pellet gains (> 60) with realistic driver energy below 10 MJ.

IFE/P6-15

Low Density Volume Ignition assisted by High-Z Shell

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Volume ignition usually takes place in high density in inertial confinement fusion (ICF). In our investigation of low power high total driving energy (LPHE) ICF approaches, we found it is possible to achieve low density volume ignition with the help of thick imploding high-Z shell. One dimensional radiation hydrodynamics simulation with a large cavity radius (a few centimeters) and low DT gas pressure ($\sim 10^{-5}$ g/cm^3) shows that, at sufficiently high driving pressure and the right amount of fusion fuel, a volume ignition like fusion process can be initiated. The ignition temperature is about 2 keV, and the fuel density of about 70 g/cm^3. The total fusion energy released is limited due to the small amount of fusion fuel. Excessive kinetic energy and extra confinement provided by the shell is essential in this type of volume ignition.

IFE/P6-16

Laser-Plasma Interaction of Petawatt-Picosecond Laser Pulses with very High Contrast Ratio

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A significant new effect was discovered with the measurement by Sauerbrey irradiating terawatt laser pulses of less than picoseconds duration producing ideally directed plane plasma acceleration where the Doppler effect resulted in accelerations of 10^{20} cm/s^2. This acceleration by the nonlinear (ponderomotive) force was predicted and numerical elaborated in exact theoretical agreement based on plasma-fluid models including the non-thermal radiation acceleration. The generated directed non-thermal plasma blocks with ion-current densities above 10^{11} Amps/cm^2 permitted a come-back of the Bobin-Chu side-on ignition of solid density DT and HB11. Further results for ^{3}He-^{3}He and p-^{7}Li are reported and problems of two dimensional fusion flames or volume ignition processes are evaluated with preparation of experimental tests.

IFE/P6-17

Resonance at ^{1}H-^{11}B Fusion applied for High Density Laser Driven Volume Ignition

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The nuclear fusion cross sections of protons and ^{11}B are interesting for re-examining the computations for laser driven fusion. This is important both for the laser driven volume ignition after a new cluster mechanism of the fuel provides very high densities in solid density targets, as well as for the new side-on ignition using Petawatt-picosecond laser pulses. Though the reaction cross sections for ^{1}H-^{11}B have been studied by several authors in the past, the special evaluation of the very pronounced resonance maximum at the reaction energy of 148 keV was given only recently by Nevins and Swain. Results are reported for the laser driven fusion energy by spherical compression where the resonance maximum was not included before. We report on new computations of the volume ignition of high density spherical compression of ^{1}H-^{11}B fusion gains showing a considerable increase due to the resonance especially at energies which are interesting for possible laser fusion.

IFE/P6-18

Coherent Tiled 4 Beam Combination by Phase Controlled Stimulated Brillouin Scattering Phase Conjugation Mirrors toward the Practical Laser Fusion Driver

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The coherent beam combination using the phase controlled stimulated Brillouin scattering phase conjugate mirrors (SBS-PCMs) is one of the promising techniques for the practical laser fusion drivers. Its ability has been demonstrated experimentally through this work. The phase fluctuations of the titled beams are less than 1/25 wavelength even when the amplifiers are inserted and operated in the beam combining system, which means that this new technique can be applied to combine the currently available lasers such as 100 J/ns/10 Hz for a real laser driver module whose output energy is greater than 5 kJ/ns/10 Hz.

IFE/P6-19

Self-Navigation of Laser Drivers on Injected IFE Direct Drive Pellets

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There are many serious obstacles complicating the classical direct drive IFE scheme - even putting in doubts its practical feasibility. Among the most serious ones is the insufficient predictability of the injected pellets' trajectories resulting from their expected interactions with remnants of previous fusion explosions due to the considered $5-10$ Hz repetition rate. This is one of the reasons why the indirect drive scheme - despite its higher demand on laser energy - seems to be currently considered a more serious IFE candidate. The corresponding hohlraum targets are by three orders of magnitude heavier compared to their direct drive counterparts thus allowing for more reliable prediction of their trajectories. In this contribution a recent progress achieved in the stimulated Brillouin scattering (SBS) phase conjugating mirror (PCM) based inertial fusion energy (IFE) approach proposed recently as an alternative to the IFE classical approach mentioned above will be presented. By taking care of automatic self-navigation of every individual laser beam on injected pellets with no need for any final optics adjustment this technology is of particular importance to the direct drive scheme. Conceptual design of one typical laser driver will be shown and its features discussed. This approach would allow for higher number of laser drivers to be employed. Operating with lower energies (~ 1 kJ – thus avoiding the optics damage caused by perpendicular SBS) such laser drivers would be easier to

design for the required repetition rate. In comparison with the earlier design an upgraded scheme was developed with the low energy illumination laser beam entering the reactor chamber through the same entrance window as used by the corresponding high energy irradiation laser beam. The pellet survival conditions in the period between its low energy illumination and subsequent high energy irradiation were studied and upper limits on the energies absorbed after their illumination for both DD and DT fuels were found. Results of experimental verification of this improved design will be reported. In these experiments for the first time a complete setup including the pellet (realized by the static steel ball) was employed. This concept of self-navigation should be, in principle, applicable also to direct drive targets with cones.

IFE/P6-20

Use of Super-Power Disk Explosive Magnetic Generators to Ignite a Target by Indirect Radiation of Z-Pinch X-Rays

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In accordance with the existing concepts the X-radiation with the energy of 10 MJ should be generated by each pinch for the time of 5 ns in the scheme of a double-pinch vacuum hohlraum to ignite a thermonuclear target. Such parameters of X-radiation can be provided at the implosion of Z-pinch by the current of more than 65 MA for the time of ~ 100 ns. Realization of such currents at the stationary facilities on the basis of the capacitor banks is still a long way in the future. The Z-machine (Sandia National Laboratories, USA), being the most powerful among the existing stationary facilities, realizes the implosion of the Z-pinch at the current of 25 MA. At the same time the developed explosive technologies allow obtaining and even exceeding the currents required for ignition. The super-power disk explosive magnetic generators (DEMG) can easily produce the currents of 300 MA for the time of $\sim 10\,\mu$s. The application of the developed technologies of the foil electrically exploded current opening switches and of the explosive closing switches allows delivering the currents to 150 MA for the time of $1 - 2\,\mu$s to the load. The paper will consider a possibility of application of the second cascade of current pulse peaking on the basis of a low-inductive electrically exploded current opening switch in order to produce the current with the amplitude to 75 MA for the time of ~ 100 ns in the Z-pinch. The paper will present the program of a step-by-step realization of such current source: a small class DEMG 250 mm in diameter, a middle class DEMG of 400 mm in diameter (an explosive counterpart of Z-machine) and, finally, a super-power DEMG 1000 mm in diameter producing the currents required for ignition.

IFE/P6-21

Stagnation of Ablated Metal Vapor in Laser Fusion Reactor with Liquid Wall

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In this paper, we will newly report 1) numerical simulation on formation of aerosols at the center where the ablated metal vapor stagnates 2) experimental result on diameter measurement of micro droplets in a simulation experiment with a Pb membrane irradiated with laser from the back-side.

In the case of KOYO-F, 10 kg of LiPb evaporates after 200 MJ fusion burn and might form lot of aerosols that would disable the nest target shot. We numerically simulate the ablation, subsequent expansion and stagnation at the center with the simulation code DECORE to discuss the influence on the chamber clearance. In this integrated simulation code, effects of condensation of a plume, phase transition from liquid to neutral gas to partially ionized plasma, absorption of energies of charged particles, equation of state, hydrodynamics, and radiation transport are included. The ablation process and subsequent expansion of vapors were calculated until they stagnated at the center. We found that there were no clusters around the leading front because the vapor expanded quickly before the growth of aerosols. After the stagnation, temperature at the stagnation point increased quickly by the PV work and no aerosols around the stagnation point were newly formed.

Another interest is hydrodynamic instabilities due to the distribution of deposited energy by alpha particles. Alpha particles release their energy at the Bragg peak. As the result, the liquid metal surface is ablated as if an $8-10$ micrometer membrane is accelerated from inside. To experimentally simulate this phenomenon, we fired a 10-micro meter-thick Pb layer on a transparent glass plate from the glass plate side with a 0.08 J/pulse, 15 ns, Nd:YAG laser. The laser energy deposited between the Pb membrane and the glass plate, which accelerated the Pb membrane. The glass plate was regarded as the residual LiPb flow. The absorbed laser energy density was estimated to 0.15 MJ/m^2 while the heat load of alpha particles in KOYO-F is 2 MJ/m^2. Accelerated vapor from the laser spot was back lighted at different timings.

Recently we measured diameter distribution in the vapor during the flight by Mie scattering method. This result clearly showed that there were lots of particles whose diameters were larger than aerosols mentioned in the previous section. We attribute formation of large particles to break of thin film by hydrodynamic instabilities. This experiment indicated that particles 10 to 100 times larger than aerosols would be formed.

IFE/P6-22

Overview on Neutronics, Safety and Radiological Protection of HiPER Facilty

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The HiPER facility is conceived to be a high repetition inertial fusion experiment dedicated to demonstrate the feasibility of the laser driven fusion. The most demanding irradiation scenario could consist in 100 MJ neutron yields, with 10 Hz repetition rate, 100 shots per burst and one burst every month, and 20 years of lifetime. The target bay should be defined to withstand this scenario. For this paper we consider that the target bay consists on the reaction chamber, a shielding for the chamber, disposable lenses, and an optical final assembly with its own shielding. The study is divided in three phases linked to different moments of the reactor lifetime: irradiation periods, cooling time between bursts and waste management after 20 years of operation. A major attention is paid in this paper to the waste management of the facility. The waste management assessment is made considering the alternatives: Near-Surface Burial, Clearance and Hands-on and Remote Recycling. The quantities identified as relevant are: Contact Dose Rate, Decay Heat and Waste Disposal Rate and Clearance indexes. Taking into account the waste management assessment, we suggest a choice for the reaction chamber material among several candidates (aluminum alloys, ferritic and austenitic steels and their reduced-activation versions). Special attention is paid to the presence of impurities. During the irradiation periods, we have studied the prompt dose rates, as well as the primary damage component, such as gas production, deposited energy, atom displacement and materials activation. Derived from this study, we calculate and propose different shielding for the reaction chamber and for the final optic assembly. It is possible also to offer an estimation of the lifetime of some pieces. In the cooling time between burst we have studied problems associated to the activation of the materials. With calculations of the residual dose rate in the whole target bay, we propose possible maintenance and replacement planning and strategies (hands-on or remote).

IFE/P6-23

The Role of the Spatial and Temporal Radiation Deposition in Inertial Fusion Chambers

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The future reactor chamber wall of HiPER will have to withstand short energy pulses (some μs long) of 5 to 20 MJ mostly in form of X-rays and charged particles every 100 to 200 ms (i.e. around 25% of the produced $20 - 100$ MJ inertial fusion energy (IFE)). This energy, if not avoided, will heat, stress, erode and damage the chamber wall and

final optics. A wise selection of wall material and chamber dimensions has to be taken in order to minimize the damage so that the facility can be fully functional during its planned lifetime.

The figure of merit for both, x-ray and charged particle threats, is the maximum fluence (J/cm^2) which the chamber components can resist without risk of failure. However, it is of paramount importance to understand that this upper limit comes determined by how these particles deposit their energy spatially and temporally on the materials. Roughly speaking, these two characteristics of the particle-matter interaction depend on the energy and temporal spectra of the incident particles and the absorption and energy transport properties of the material.

Calculations using different approaches for energy deposition on tungsten will be presented, showing the importance of the spatial and temporal deposited energy profile in Inertial Fusion Tungsten armor material.

IFE/P6-24

Modeling Advanced Materials for Nuclear Fusion Technology

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A particular challenge in material sciences is the design and synthesis of molecular structures with desired physical properties. Atomistic simulations using MD techniques play a crucial role in this task.

Here we present, at the atomic scale, first results and intended general methodology in the study of radiation damage and non-linear wave propagation/response to high shock loading for different advanced materials. We have studied materials ranging from single crystal structures to more complex nanostructures (nanocrystals) or nano-foams. We report non-equilibrium Molecular Dynamics simulations, using empirical MEAM potentials, able to cover long time scales (in a order comparable to experiments).

We would like to raise the interest over these new nanostructured materials, such as nanocrystals, and porous metals (metallic foams), since are perfect candidates to take part in different critical components in a Fusion reactor. For instance, the National Ignition Facility (NIF) at Lawrence Livermore National Laboratory will require polycrystalline ignition targets. On the other hand, metallic foams appear as a possible alternative in current designs for targets in the HiPER initiative.

IFE/P6-25

Study of Diffusion and Retention of Light Species (H and He) in Pure W and W-based Materials

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We report on diffusion and retention (depth profiling) experiments of light atoms (H and He) in co-implanted and in post-implanted annealed pure W and W-based materials to be used in future fusion reactors. The understanding of these phenomena is crucial in order to design armor and structural materials because nucleation of these gases in defects may lead to bubble formation and subsequent deterioration of mechanical properties. This ion bombardment represents one of the most significant differences between fusion and fission technologies.

The microstructure of the samples was investigated prior to and after the different irradiations by means of X-ray diffraction (XRD) and scanning electron microscopy (SEM). The chemical composition of the samples was analyzed prior to implantation by means of energy dispersive spectroscopy (EDS) and particle induced X-ray emission (PIXE). He and H depth profiles were characterized by elastic recoil detection (ERDA), and nuclear reaction analysis (NRA). Mechanical tests were performed to study the effects of ion bombardment on the mechanical properties.

In this contribution, the influence of the material composition and microstructure on the diffusion and retention of light species is discussed on the basis of existing models.

IFE/P6-26

Target Fabrication for the National Ignition Facility and for Inertial Fusion Energy

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Targets for Inertial Confinement Fusion (ICF) experiments are being manufactured for the National Ignition Campaign (NIC) utilizing the National Ignition Facility (NIF) at Lawrence Livermore National Laboratory. These targets for indirect drive ICF consist of a spherical ablator shell and its concentric cryogenic deuterium-tritium fuel layer placed in the center of a cylindrical hohlraum. There are many requirements, from capsule roughness and sphericity to hohlraum length and thickness that target fabrication methods and processes need to meet in a production environment. This paper describes the state-of-the-art materials and technologies that have been developed to produce these targets at

high yiel, and the prospects for achieving targets at high rate and low cost for inertial fusion energy (IFE).

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IFE/P6-27

Recent Development of Target Fabrication and Fuel Layering Technique for FIREX project

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Several key issues for fabrication and fuel layering technique of cryogenic target for Fast Ignition Realization Experiment (FIREX) project have been developed. From the viewpoint of enhancement of coupling efficiency on the fast heating and improvement of the implosion performance some advanced modifications for the target design have been proposed. Fabrication of low-density plastic shell and their mass production technique using new devices have been developed. Furthermore, for the fuel layering, the new technique of conical laser guide heating has been proposed to realize a uniformly fuel layer.

IFE/P6-28

Effect of Neutron Irradiation on the Characteristics of Laser Produced Plasma

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We investigate the effect of structural defect levels in crystal structure of targets on the threshold and morphology of destruction of the targets during the interaction of intense laser radiation with target surface. We also study the characteristics of plasma ion formation from the target. We show that structural changes in irradiated Al and W targets have an influence not only on the efficiency of the process of material evaporation and emission of plasma bunch, but also on the efficiency of ionization and recombination processes, which takes place in the plasma bunch on the stage of formation and expansion.

ITR
ITER Activities

ITR — Oral

ITR/1-1

Preparations of the ITER Vacuum Vessel Construction

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ITER vacuum vessel (VV) consists of nine 40 degree sectors and ports will be procured by EU, KO, RF and India. Procurement Arrangements for the vacuum vessel have been signed by the relevant Domestic Agencies. Each DA will make the fabrication contract with industries within 2010. After the preliminary approval of VV design by ANB, the related DAs will start the fabrication works based on this design. The R&D for set up the fabrication procedures to meet the demanding tolerance and inspection requirements is the first activity and the results of this R&D from each DA are presented.

ITR/1-2

Optimizing the ITER Heating and Current Drive Mix

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ITER will be heated in the first stage by 20/33/20 MW for ECH, NBI and ICH. All three systems heat the plasma effectively. The modelling of EC current drive efficiency gam-E requires parallel momentum conservation. The two momentum conserving codes CQL-3D and TRAVIS show consistency. Modelling of the NBCD efficiency, gam-N, was done with OFMC and NUEBEAM. The benchmarks yield differences of about 25%. Both ECH and NBI show similarly high gamma-values in the core. gam-N stays fairly constant with radius qualifying NBCD for global CD; γ_E drops sharply toward the edge. The following observations were made: With sufficient edge stability $Q = 10$ is possible with NBI or ECH. ICH heats selectively the ions and increases fusion gain by $\delta Q \sim 1$. A flat-top time of 350 s can be reached with an early H-mode l_i. The $Q = 5$, quasi- steady-state condition requires discharges with $f_{bs} > 50\%$ and strong current drive. The various tasks have to be split within the heating methods: ECCD deposited off-axis can initiate reversed shear scenarios but not effectively drive the current. This can only be achieved by NBI. With

ECCD alone, $Q = 5$ cannot be reached independent of the heating power. With beams, inductive discharges with $f_{ni} > 80\%$ and $Q \sim 5$ can be maintained for 3000 s. Prototype testing is underway for gyrotrons, transmission line and both launchers. The ITER-like IC antenna of JET has validated the concept of multiple short straps on ELMy plasmas. A second antenna has been included to ensure the availability of sufficient power. The development risks of NBI at 1 MeV is being mitigated with a series of test facilities. The risks can be further reduced by operating at 0.85 MeV necessitating increased source current density. Also DEMO will need NBI for steady state operation. In summary, a mix of heating and CD systems provides the necessary actuators to respond in a flexible way. The $Q = 5$ steady-state prospects should be enhanced with the addition of 20 MW LHCD and 20 MW ECH for q-profile control and a higher safety margin in reaching the H-mode. The ECH extension would also serve as back-up in case other systems do not reach the specified powers.

ITR/1-3

ITER Test Blanket Module Error Field Simulation Experiments at DIII-D

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Comprehensive DIII-D experiments investigated the effects of ferromagnetic error fields similar to those expected from proposed ITER Test Blanket Modules (TBMs). Studied effects included: plasma confinement; L-H transition; edge localized mode (ELM) suppression by resonant magnetic perturbations (RMPs); locked modes; energetic particle losses; and more. DIII-D experiments used a 3-coil mockup of 2 magnetized ITER TBMs in an ITER equatorial port. The largest and most prevalent effect was slowing of plasma toroidal rotation ν across the entire radial profile by as much as $\delta\nu/\nu_0 \approx 50\%$. The slowing is attributed to non-resonant braking by the TBM mockup magnetic field B. There was no sign of braking by resonant magnetic harmonics. These results are consistent with the near absence of resonant helical harmonics in the TBM magnetic spectrum. Changes in global confinement ($\delta n/n_0$, $\delta\beta/\beta_0$, $\delta H_{98}/H_{980}$) were $\sim$ 3 times smaller than $\delta\nu/\nu_0$. The results did not depend strongly on whether b was imposed by mockup coil current variations or mockup-to-plasma distance changes. TBM effects increased with increasing

β_N and were smaller in L-mode. H-mode power threshold rose $\leq$15% within the uncertainty of the measurements. Mockup field increased the plasma sensitivity to mode locking by known, deliberately applied $n = 1$ error field proxies (n = toroidal harmonic number). The standard DIII–D locked mode tolerance at low β was recovered by the simple expedient of a new empirical compensation of the total $n = 1$ error. At high β the increased sensitivity to locking was due to TBM braking torque slowing the plasma. Velocity reduction may be the primary cause of the confinement reductions. Quantifying TBM effects on ELM suppression by $n = 3$ RMPs was difficult, because small TBM B had little effect on ELMs but larger B ($\sim$ 3 times one ITER port's error) slowed the rotation of neoclassical tearing modes in the high-β test plasmas, which often produced a back-transition to L-mode. It may be possible to extrapolate a number of these results to an ITER with 2 TBMs in a single port, but extrapolating to 6 TBMs in 3 ports is less clear. Effects related to low-n TBM harmonics may be correctible by conventional error field correcction.

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ITR/1-4

ITER ELM Control Requirements, ELM Control Schemes and Required R&D

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ITER operation in high QDT scenarios (inductive, hybrid and steady-state) relies on the achievement of the H-mode confinement regime with $H_{98} = 1$ and a significant plasma pressure gradient at the plasma edge which is expected to lead to the quasi-periodic triggering of ELMs. Operation of ITER with H-mode plasmas is also foreseen during the non-active (H and He) and DD operation. This will allow the study of the features of plasmas in this confinement regime at the ITER scale and to develop scenarios for DT operation in ITER during these pre-DT phases, including ELM control schemes. Extrapolation of measurements of ELM energy losses and energy fluxes to plasma facing components (PFCs) in present devices to ITER indicate that for naturally occurring ELMs (or "uncontrolled"), the energy fluxes associated with ELMs will produce an unacceptably low plasma facing component lifetime during DT operation because of excessive erosion and/or superficial surface damage. Requirements to control the energy loss by repetitive ELMs in ITER have thus been defined for ITER on the basis of experimental results to avoid enhanced ELM erosion associated with surface overheating of PFCs.

To achieve these ELM control requirements, it is planned to equip ITER with two major systems: a) in-vessel coils to produce a resonant edge magnetic perturbation with the same characteristics as those found to lead to complete ELM suppression in DIII-D and b) pellet injectors able to reduce the ELM energy loss by the controlled triggering of ELMs with the edge perturbation induced by the pellets whose injection frequency can be adjusted to the requirements. Besides these two baseline systems other control schemes are being considered for ITER (such as vertical plasma position oscillation), but the feasibility and compatibility with other design requirements of these alternative control

schemes remains to be assessed. Although the physics basis for both the ELM control requirements and ELM control schemes in ITER are reasonably well established, many important issues remain to be studied with respect to their application in ITER, which are being addressed both by experiments at the ITER Members' facilities and detailed modelling for ITER and whose major results will be summarised in the paper together with the open issues where further R&D is required.

ITR/1-5

Development of Advanced Inductive Scenarios for ITER

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Since its inception in 2002, the International Tokamak Physics Activity topical group on Integrated Operational Scenarios (IOS) has coordinated experimental and modeling activity on the development of advanced inductive scenarios for applications in the ITER tokamak. This report documents the present status of the physics basis and the prospects for applications in ITER. The key findings of this research activity are: 1) inductive scenarios capable of higher normalized pressure ($\beta_N \geq 2.5$) than the ITER baseline scenario ($\beta_N = 1.8$) with normalized confinement at or above the standard H-mode scaling have been established under stationary conditions on the four largest diverted tokamaks (AUG, DIII-D, JET, JT-60U); 2) the parameter range where high performance is achieved is broad in q_{95} and density (normalized to the empirical density limit) and is common to all four tokamaks; 3) MHD modes can play a key role in reaching stationary high performance, but also define the stability and confinement limits; 4) results from individual machines with unique capabilities for varying rotation, current drive, and heating sources facilitate more realistic projections for ITER performance; 5) coordinated experiments have yielded clearer measurements of the normalized gyroradius scaling that dominates the projection to ITER; and 6) coordinated modeling activity supports the present research by clarifying the most significant uncertainties in the projections to ITER. Studies extending previous work on pedestal characterization, radiative divertor operation, and

edge localized mode (ELM) suppression to advanced inductive scenarios have also been coordinated through the IOS group.

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ITR/1-6

Disruption Mitigation in ITER

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Success of the ITER program depends critically on the development of robust and reliable techniques for disruption mitigation. If unmitigated, disruptions at full plasma current and high stored energy will result in intolerably high thermal loads on plasma- facing components, large forces on plasma-facing conducting structures such as blanket modules and the vacuum vessel, and the generation of massive runaway electron currents which will readily lead to first wall melting. Although ITER is designed to withstand a certain number of full scale disruptions, the consequences of even a few unmitigated events will be extremely serious in terms of component lifetime. The present paper describes possible approaches to Disruption Mitigation Systems (DMS) on ITER, together with progress in the development program of these systems, including experimental tests of the mitigation techniques on present tokamaks and in the laboratory. Emphasis is also placed on the limitations imposed on DMS by the ITER design and mode of operation.

ITR/2-1

The ITER Magnet Systems: Progress on Construction

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Construction of the ITER magnet systems has been started at the end of 2007 following the signature of the first Procurement Arrangements for the Toroidal Field (TF) conductors. Six ITER Parties are involved in the share of the ITER magnet components and, to date, more than 15 Procurement Arrangements between the ITER Organization and the Parties have been signed. Substantial progress towards full scale construction has been

achieved with the placement of the first large manufacturing contracts, the production of several tens of tons of advanced Nb3Sn and NbTi strand, and the set-up of large cabling and jacketing facilities across six ITER Parties. In most cases, the detailed design of the coils and support structures has also been finalized and reviewed. The qualification of the fabrication processes for the Toroidal Field (TF) coils and Poloidal Field (PF) coils has been initiated. The preliminary design of the Central Solenoid (CS) and the detailed design of the Correction Coils (CCs) have been carried out, as well as for the TF gravity supports and clamps of the PF coils. The manufacture of prototypes of the feeder lines and current leads has been started, while ITER is in charge of the qualification and procurement of the required magnet instrumentation.

ITR/2-2

Review of the ITER Fuel Cycle

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The ITER Fuel Cycle consists of three major sections which are Vacuum, the Tritium Plant and Fuelling and Wall Conditioning. It can be further divided into 18 systems as outlined in Figure 1. In general the Fuel Cycle supplies the fuel particles and impurities into the tokamak for stable plasma operation and processes all the exhaust gases from the Torus. The ITER Fuel Cycle must be capable of delivering, exhausting and purifying fuel particles (H_2, D_2 and DT) at average and peak rates of 200 Pa·m^3 s^{-1} and 400 Pa·m^3 s^{-1} respectively, as well as impurities such Ne, Ar and N_2 with average and peak rates of 10 Pa·m^3 s^{-1} and 100 Pa·m^3 s^{-1}. These rates are orders of magnitude higher than have been done previously in industrial applications with fluids containing tritium. This paper describes the major interfaces between these systems, the overall design status, schedule strategy and challenges associated with design of the ITER Fuel Cycle Systems.

ITER is a fusion tokamak, being fully designed for deuterium / tritium operation. The plasma will be fuelled in the forms of hydrogenic ice pellet injection and gas puffing. The former mainly provides plasma density control and ELM frequency control; the latter for plasma detachment control and enhancement of radiative cooling in diverter region. The Fuelling system also provides hydrogen and deuterium to the diagnostics and heating neutral beam injectors respectively. In addition to the proven techniques for plasma density control it will be possible to control the neutral particle exhaust on ITER by varying the opening of the torus cryopumps. The ITER vacuum pumping systems are also used for initial evacuation and continuous maintenance of the required conditions in the torus. This system consists of a high vacuum cryo-pumping part and a roughing system, which discharge to the Tritium Plant systems. The latter supplies deuterium and tritium from external sources and treats all tritiated fluids from ITER operation through Tokamak Exhaust Processing, Isotope Separation and Storage and Delivery Systems to remove and recover deuterium and tritium for refuelling.

ITR/2-3

Power Handling in ITER: Divertor and Blanket Design and R&D

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One of the most technically challenging components of the ITER machine are those directly facing the thermonuclear plasma. The plasma-facing components (PFCs), include the Divertor and the Blanket, and consist of a plasma facing material (the "armour") mounted onto a heat sink and supported by a structural/shielding material. The development of suitable technologies for the power handling of the stationary thermal loads has required an unprecedented R&D effort by all the concerned Domestic Agencies over the last two decades.

The main function of the divertor is exhausting part of the plasma thermal power (including the alpha particle power), as well as minimizing the helium and impurity content in the plasma.

The ITER divertor consists of 54 cassette assemblies which are inserted radially through three lower level ports and moved toroidally before being locked into position. Each cassette assembly includes one Cassette Body and three PFCs, namely the inner and outer Vertical Targets, and the Dome. The inner and outer targets are the PFCs, which in their lower parts directly interact with the SOL and in their upper parts act as baffles for the neutrals.

The main functions of the blanket system are: (1) to exhaust part of the plasma power, (2) to contribute in providing neutron shielding to superconducting coils and (3) to provide limiting surfaces that define the plasma boundary during startup and shutdown.

The system comprises two different sub-systems that cover the ITER inner vessel wall. The wall-mounted blanket modules covering 620 m^2 and the port-mounted blanket modules covering 40 m^2. The wall mounted blanket modules (BMs) consist of two major components, a plasma-facing First Wall panel, with a beryllium armour, supported by a Shield Block.

The required remote handling, the extremely high electromagnetic and thermal loads, the number of physical interfaces with the rest of the machine and with the diagnostics as well as the direct interaction with the plasma, make the identification of a viable design solution of the ITER PFCs one of the most challenging design and R&D efforts of the entire machine. This paper summarizes the status of the Divertor and the Blanket, the related R&D effort, and gives an outline of the construction schedule.

ITR/2-4Rb

1 MV Holding and Beam Optics in a Multi-Aperture Multi-Grid Accelerator for ITER NBI

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In a negative ion accelerator utilized in the ITER neutral beam (NB) system, generation of 1 MeV, 40 A (200 A/m^2) deuterium negative ion (D^-) beams is required for 3600 s. To meet the requirement, a negative ion accelerator, called a "MeV accelerator", has been developed at Japan Atomic Energy Agency (JAEA). The MeV accelerator is a five stage Multi-Aperture Multi-Grid (MAMuG) accelerator, which was chosen as a baseline accelerator design for the ITER NB. A target of the MeV accelerator is to accelerate 0.5 A (200 A/m^2) H^- ion beam at the energy of 1 MeV for several tens seconds. After successes of beam acceleration at 836 keV, 140 A/m^2, 0.2 s presented at the previous conference of this series, present R&D are dedicated to increase the pulse length. For stable MeV-class beam acceleration, voltage holding studies were performed. Although structure and grid area are different, voltage holding capability of the MeV accelerator and JT-60U accelerator shows the same dependence on the gap length. Moreover, sustainable voltage of these accelerators at JAEA is a half of that in the small electrode used for design of these accelerators. The result indicates that the gap lengths in the present MeV accelerator (minimum gap; 72 mm) are considered to be marginal for stable voltage holding. Then, the gap length was extended to 100 mm so as to hold 200 kV stably (1 MV for five stages) including the margin of 20%. As a result, the MeV accelerator achieved stable voltage holding of 1 MV without gas feeding (base pressure; 2×10^{-4} Pa) for 4000 s. In the beam acceleration test, it was observed that relatively high heat loads on acceleration grids were occurred. So as to investigate the cause of the heat loads, a 3D multi beamlet analysis was applied to investigate the beam deflection due to magnetic field and space charge repulsion.

ITR/2-5Rb

Development of High Power Gyrotrons and EC Technologies for ITER

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Recent progress on the development of a high power 170 GHz gyrotron and EC (electron cyclotron) technologies in Japan Atomic Energy Agency (JAEA) is presented. After the demonstration of ITER basic performance on the 170 GHz gyrotron, a repetitive operation test, a long pulse power modulation for NTM (neo classical tearing mode) control, and a dual frequency gyrotron development have been conducted. In the repetition test, the 800kW/600s shot at the electrical efficiency of 52-57% with the depressed collector

is reliably repeated with an interval of $\sim$ 20 min in 10 days as a simulation of ITER operation. The 72 shots of 88 shots successfully achieved the 600 sec oscillation. As for the power modulation, 5 kHz full power modulation was demonstrated firstly in CW mode with a full beam current modulation. In a development of the dual frequency gyrotron, the 1.3 MW oscillations at 170GHz and 136GHz were successfully demonstrated by changing the applied magnetic field. Using the gyrotron, high power RF transmission test and RF radiation test have been conducted with a mock-up set-up of ITER EC system.

ITR/2-5Rc

2.2 MW Operation of the European Coaxial-Cavity Pre-Prototype Gyrotron for ITER

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A 2 MW, CW, 170 GHz coaxial-cavity gyrotron for electron cyclotron heating and current drive in the International Thermonuclear Experimental Reactor (ITER) is under development within an European Gyrotron Consortium (EGYC*). To support the development of the industrial prototype of a CW gyrotron, a short pulse tube (pre-prototype) is used at KIT (former FZK) for experimental verification of the design of critical components, such as electron gun, beam tunnel, cavity and quasi-optical (q.o.) RF-output coupler. Significant progress was achieved recently. In particular, RF output power of up to 2.2 MW with 30% output efficiency has been obtained in single-mode operation at 170 GHz. Furthermore, an excellent quality of the RF output beam with $\sim$ 96% fundamental Gaussian mode content has been obtained by using a new quasi-optical RF output system. The verification of these results with computer simulations will be presented.

ITR/P1 — Poster Session

ITR/P1-01

Neutral Particle Analysis on ITER: Present Status and Prospects

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In this report we present the abilities of the Neutral Particle Analysis (NPA) measurements for ITER steady-state (SS) operation scenario in respect to the main objective of the NPA diagnostic on ITER, which is to measure the DT fuel composition of the fusion plasma.

We analyze the physical basis for measuring of the neutral particle fluxes emitted by ITER plasma and describe the instrumentation for that which is being developed in Ioffe Institute (St.Petersburg, Russia). Possible mechanisms of the neutralization of the hydrogen and helium ions in the thermal and supra-thermal energy ranges are considered. Numerical simulation results of the neutral fluxes produced by the neutralization processes of the bulk thermal deuterium and tritium ions, Neutral Heating Beam particles, "knock-on" deuterium and tritium ions and the confined alpha particles are presented.

NPA tandem arrangement consisting of Low Energy Neutral Particle Analyzer (LENPA) and High Energy Neutral Particle Analyzer (HENPA) is described. The analyzers provide the measurements in thermal ($10 - 200$ keV) and supra-thermal ($0.1 - 4$ MeV) ranges respectively.

Calculation of the counting rates in the analyzer energy channels shows that the NPA system will be able to measure D/T ratio both in ITER plasma core ($r < 0.4a$) by measuring of the neutralized "knock-on" deuterons and tritons and in plasma medium region ($r > 0.4a$) by measuring of the neutralized thermal deuterons and tritons. A possibility to measure the flux of the neutralized confined alphas is also discussed in the report.

ITR/P1-02

R&D ITPA Activities in Support of Optimizing ITER Diagnostic Performance

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While much progress has been accomplished in the detailed design and integration of the ITER diagnostics, a few generic issues related to ITER environment are still outstanding. ITER diagnostics will experience harsh conditions including but not limited to large radiation flux and fluence, large neutral (atomic) flux, large heat flux at the first wall, reduced access, difficult calibration schemes and long pulses. Addressing these issues and how they can affect diagnostic performance are the focus of the Topical Group's High Priority Items, which are: 1) Development of methods of measuring the energy and density distribution of escaping alpha particles; 2) Assessment of the calibration strategy and calibration source strength needed for the neutron diagnostics; 3) Determination of the life-time of plasma facing mirrors used in optical systems; 4) Assessment of techniques for measurement of dust and erosion; 5) Assessment of impacts of in-vessel wall reflections on optical diagnostics; and 6) Assessment of the measurement requirements for plasma initiation and identification of potential gaps in planned measurement techniques. This paper will describe these issues and present the current status in the development of solutions and accompanied R&D. Other topics, such as protection of diagnostics in presence of large microwave stray power will also be discussed.

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ITR/P1-03

Development of Gamma-Ray Diagnostics for ITER

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Gamma-ray spectrometry provides valuable information on fast ions (H, D, T, ^{3}He) and fusion products (p, t, alpha-particles) in tokamaks. It identifies fast ions distinguishing different species and gives their energy and spatial distributions by means of tomografic reconstruction of measured gamma-ray emission profiles. In ITER, the horizontal profile gamma-camera has been designed. The feasibility study for the vertical one is in progress. The absence of a suitable vertical port in ITER makes this rather complicated. One of the options is to use the divertor port for the vertical viewpoint implementation. Strong

magnetic field ~ 2 T in the divertor area makes the using of conventional multi-dynode photomultipliers impossible for neutron and gamma detection systems. It was proposed using micro-channel plate (MCP) photomultipliers instead. In this paper the results on the MCP gamma-spectrometer performance study are presented. The high-speed pulse height analysis technique developed in the Ioffe Institute allows tracing of the MCP-photomultiplier gain changing. These tests demonstrated feasibility of using the MCP photomultiplier based detectors for gamma-ray and neutron spectrometry in the divertor port zone of ITER.

ITR/P1-04

Status of Design and R&D for the Korean ITER Diagnostic Systems

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Development and implementation of the ITER diagnostic systems have major challenges because the environment is much harsher and the requirements for plasma and first wall measurements are much harder than present devices. About 40 diagnostic systems are designed to be implemented to provide accurate and reliable measurement for a wide range of plasma parameters, in total about 45 parameters. All the ITER members are contributing diagnostics of which the ITER Korea (Korean domestic agency, KODA) will procure one diagnostic upper port plug and three diagnostic systems of a neutron activation system (NAS) and, main and divertor vacuum ultraviolet (VUV) spectrometer systems. Implementation activities including R&D on ITER diagnostics in Korea started shortly after procurement assignment. Currently, the ITER Korea is conducting detailed design and engineering work to resolve the outstanding technical issues and to finalize the technical specifications and also participating in joint activities for development of technical documents of the procurement arrangement (PA), which include measurement requirements, functional block diagrams, procurement schedule tables, and engineering annexes. In this paper, the status of design and results of R&D for the Korean ITER diagnostic systems will be described.

ITR/P1-05

Mirrors for ITER Diagnostics: New R&D Developments, Assessment of the Mirror Lifetime and Impact of the Mirror Failure on ITER Performance

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All optical and laser-based diagnostics in ITER will use mirrors as the first plasma-viewing component. In the harsh radiation and particle environment, optical properties of mirrors will degrade leading to the deteriorated operation of the entire respective mirror-based diagnostics. Prioritized R&D program is underway to ensure the most durable high-performance mirror solution. This activity is ongoing under the coordination of the Specialists Working Group on First Mirrors established under the auspices of the ITPA Topical Group on Diagnostics. An analysis of newest developments in the area of first mirrors is presented in this contribution including an overview of the recent results of the mirror test at JET, progress on the laser and plasma cleaning of diagnostic mirrors, a new insight to predictive modeling and the recent technological developments on mirror manufacture. Based on these developments, an analytical assessment of the mirror lifetime is given. The combination of the proper mirror material, active and passive techniques for deposition mitigation and mirror cleaning may gain orders of magnitude increase of the mirror lifetime. The degradation of any particular diagnostic system due to mirror failure will have consequences to the overall operation and performance of ITER. The impact of a given system varies greatly depending on its primary purpose, the related measurement requirements, the local environment around the mirror, and the availability of secondary diagnostics to replace or supplement the primary one.

ITR/P1-06

First Optics in ITER: Material Choice and Deposition Prevention/Cleaning Techniques

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In recent years, behaviour of different mirrors have been studied in experiments modelling the behaviour of optical diagnostic mirrors inside ITER. These mirrors have many requirements to meet. They should be resistant to sputtering by fast ions and neutrals, to temperature fluctuations, to neutron, gamma, X-Ray and UV irradiation, robust against deposition of impurities and finally they should be made of nonreactive materials. Actually, the mirrors in ITER can be divided into two types depending on whether they are expected to operate in the erosion-dominated or deposition-dominated conditions. Mirrors to be used under deposition-dominated conditions will undergo intensive contamination with products of the plasma-induced erosion of first-wall components and divertor tiles. The durability of mirrors in such an environment depends on their design and the mirror material. Our efforts are focused on two main directions:

1. design of mirrors which can withstand severe environmental conditions in ITER,
2. development of prevention/cleaning techniques.

ITR/P1-07

Overview of High Priority ITER Diagnostic Systems Status

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Diagnostics are a critical part of an operational ITER. They will provide all the information on the performance and performance optimisation. At this stage of the tokamak device development, it is imperative that access, boundaries and requirements are well established. This will allow a smooth path towards implementation of the measurement systems as well as minimising effort and ultimately cost. The ITER device will be distinguished by being the largest magnetic fusion device ever built and the first to be licensed as a nuclear facility. It will have operating conditions that have hitherto been unexplored. These include a combination of fusion power and long pulse operation. The combination of the above creates a set of conditions that place many diagnostics in a completely new regime. While optical systems have to access the plasma, they have to also be shielded to avoid direct particulate contamination as well as neutron and mechanical stresses. As well as this, minimisation of activation is required outside the vacuum boundary. These requirements combine to provide a challenging base from which to design the systems. Much R&D is ongoing to provide a sound basis for many of the designs. Specifically a report on upper port engineering, magnetic and infrared systems as well as integration will be presented.

ITR/P1-08

Simulations of Material Damage and High Energy Fluxes to ITER Divertor and First Wall during Transients and Runaway Electron Loads

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The anticipated regime of the tokamak ITER is the H-mode in which the repetitive outbreaks of the edge-localized mode (ELM) produce plasma fluxes which determine the erosion rate and the lifetime of PFCs. The disruptions also reduce the PFC lifetime, despite of mitigation measures such as the massive gas injection (MGI), in particular because of high heat fluxes by runaway electrons and the radiation flush. The lost plasma dumped mainly into the scrape-off layer (SOL) produces surface erosion by sputtering, melting, splashing, cracking and vaporization.

The expected transient heat fluxes on the PFCs are: Type I ELM $0.5 - 4$ MJ/m^2 on the timescale $0.3 - 0.6$ ms, thermal quench flux $2 - 13$ MJ/m^2 in $1 - 3$ ms. Mitigated disruption radiative flux $0.1 - 2$ MJ/m^2 in $2 - 5$ ms, and the runaway flux more than 10 MJ/m^2 on the timescale $10 - 100$ ms. In ITER the CFC and tungsten macrobrush armour as PFCs for the divertor and the dome, and beryllium macrobrushes for the first wall (FW) are foreseen. The fluid motion in a thin molten layer of W and Be during transients may produce melt splashing and thus dust emission by droplets.

The expected erosion of ITER PFCs can be properly estimated by numerical simulations validated against erosion experiments at the plasma gun facilities QSPA-T, MK-200UG and QSPA-Kh50. The measured material erosion was used to validate the melt dynamics code MEMOS and the thermomechanic code PEGASUS that were then applied to model the erosion of ITER PFCs under the anticipated transient loads. The results of experiments carried out at QSPA-T allowed validation of numerical model for the melt splashing based on Kelvin-Helmholtz- and Rayleigh-Taylor instabilities. The crack formation at W surface was modeled using the code PEGASUS and validated against the experiments carried out at QSPA-Kh50. The models were applied for simulations of PFCs damage under expected ITER-like scenarios. Numerical simulations under radiation and runaway electron impact with code MEMOS performed for expected ITER like heat loads demonstrated importance of evaporation from the armour surface which significantly decreases the melt sickness.

The paper will also describe the integrated tokamak simulations developed in KIT for transient loads. Latest modelling results of TOKES on the radiation energy distribution over the vessel surface during MGI in ITER are presented.

ITR/P1-09

Testing of ITER-Class ECH Transmission Line Components at the JAEA Radio-Frequency Test Stand

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Testing of prototypical ITER ECH transmission line components at representative ITER conditions has been carried out at the JAEA RF test stand (RFTS). Based on these tests, it appears that the ITER requirement for 2 MW cw operation with high transmission efficiency is achievable. The tests showed that improvement in the polarizer miter bend mirrors is needed and that it is critical to have accurate alignment of the gyrotron output into the waveguide. Progress in both areas is being made. GA components tested in 2008 included polarizer miter bends, waveguides, very low diffraction loss miter bends (LDLMBs) and a DC break. Components to be tested in 2010 include a waveguide switch, waveguide water cooling bars, a sliding waveguide joint and polarizer miter bends with improved grooved mirrors. For the 2008 tests, JAEA measurements and analyses show that the beam entering the test section had an HE_{11} content of 66%, with the rest of the power in LP_{11} and other high order modes. JAEA subsequently improved the HE_{11} mode purity to 93%, which is approaching the ITER requirement of $> 95\%$. Two polarizer miter bends separated by a 2-m length of waveguide were tested. Measurements of the DC break showed that the temperature increase of the outer surface of the ceramic insulator was about 8°C (equivalent to 27°C for 2 MW transmission) – this increase is still less than a safe maximum increase of 50°C. For pure HE_{11}, the calculated temperature increase is about 10°C for 2 MW transmission. The polarizer mirrors had losses 1.7 to 2.9 times the theoretical losses at room temperature. This high loss is attributed to a resistive recast layer on the surface of the copper grooves created by the wire-EDM process. The polarizers were returned to GA, their mirror surfaces were re-machined using conventional NC machining, and they will be retested at the JAEA RFTS. The LDLMB tests showed that mirror losses, especially on the downstream LDLMB (0.21%), were close to the theoretical prediction (0.16% at room temperature). Losses in adjacent waveguides were greater than expected for pure HE_{11} transmission because of higher ohmic and mode conversion losses of the high order modes. A prototype sliding joint waveguide was fabricated and will be tested in 2010 at the JAEA RFTS.

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ITR/P1-10

An Overview of the ITER EC H&CD System and Functional Capabilities

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A 24 MW CW Electron Cyclotron Heating and Current Drive (EC H&CD) system operating at 170 GHz is to be installed for the ITER tokamak. The EC system will represent a large step forward in the use of microwave systems for plasma heating for fusion applications; present day systems are operating in relatively short pulses ($\leq$ 10 s) and installed power levels of less than 4.5 MW. The magnitude of the ITER system necessitates a worldwide collaboration. This is also reflected in the EC system that is comprised of the power supplies, sources, transmission line and launchers. A partnership between Europe, India, Japan, Russian Federation, United States and the ITER organization is formed to collaborate on design and R&D activities leading to the procurement, installation, commissioning and operation of this system. The functional capabilities of the EC system strongly depend on the focusing and access range provided by the two launcher types: one in the equatorial (EL) ad four in the upper (UL) ports. The original baseline design (prior to 2007) partitioned the physics applications between the two launchers based on regions in the plasma. The EL accessing inside of $\rho_T \leq 0.5$ for applications associated with central heating, current drive and sawtooth control. The UL accessing the region where NTMs are expected to occur between $0.55 \leq \rho_T \leq 0.85$. The combination of two

launchers could not achieve the desired accessibility. The design Teams have modified the launcher designs increasing the access range and avoiding regions of non-accessibility. The physics applications have been repartitioned between the two launchers, dividing the functional requirements based on the need for broad (EL) or narrow (UL) deposition profiles. The EL would still be used for central heating and current drive applications, while the UL used for the control of both the NTM and sawtooth instabilities. In addition, one third of the EL beams are to provide counter current drive to decouple the heating and current drive contributions when depositing power centrally. The aim of this paper is to provide a brief review of the design improvements incorporated following the ITER design review of 2007 and the increased functional capabilities.

ITR/P1-11

On Maximizing the ICRF Antenna Loading for ITER Plasmas

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ICRF plasma heating relies on the propagation of the fast wave, evanescent until it reaches the cut-off density (10^{18} m $^{-3}$ range). The maximum achievable coupled power will varies to a first approximation linearly with the coupling resistance itself decreasing exponentially with the antenna – cut-off density distance. As the SOL density in ITER, is expected to vary widely, gas injection was proposed to minimise the sensitivity of the ICRF coupling to the edge density and joint experiments coordinated by the ITPA were performed to characterise further this method. Although the increase in SOL density due to gas injection, always led to a coupling improvement, for a given level of gas injected significant differences in the coupling increase and effect on the confinement were measured depending of the plasma pumping and recycling properties of the device and plasma configuration used. Nevertheless, regimes of operation were identified where the coupling was substantially improved without significant loss of confinement. Moreover, on JET, gas injection from the main chamber led to a larger coupling improvement than injection from the divertor and depending of the plasma shapes, gas injection from inlet magnetically connected to the antennas resulted in a significantly higher coupling. AUG and DIII-D data also illustrated that gas injection from pipes located on the antenna limiters or adjacent to the antenna gave the highest coupling improvement. The effect of gas injection on phenomena related to RF sheaths was also studied. Both in JET and in AUG, the metallic impurities concentrations, which generally increase with the ICRF power, were reduced when increasing the D fuelling and on Tore Supra, the importance of a trade-off between coupling improvement and heat-loads, was shown. These results indicate that injecting gas in ITER from gas inlet either adjacent or at least magnetically connected to the antennas could be an ideal tool to tailor the edge density in front of the ICRF antenna and maximise the ICRF power output over the widest range of plasma parameters. They emphasise the critical importance of having density measurements in front of the ITER ICRF antennas and the urgent need to develop modelling of the ITER far-SOL density modification during gas injection, including enhancement of the ionisation by antenna near-field.

ITR/P1-12

Prototype Manufacturing and Testing of Components of the ECH Upper Launcher for ITER

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Besides from heating the plasma the Electron Cyclotron Heating (ECH) Launcher at ITER has to provide local current drive in order to control plasma instabilities with the main focus on the stabilisation of neoclassical tearing modes (NTMs) and the sawtooth instability. The injection of 20 MW mm-wave power at 170 GHz is guided into the plasma by a mm-wave system which is installed in 4 of the 18 upper port plugs of the ITER vacuum vessel. The structural system of these so-called Upper Launchers, housing the mm-wave system, consists of two units, namely the blanket shield module (BSM) which forms the plasma facing component and the launcher main structure with removable internal shield blocks. It has to be designed for rough operation conditions as nuclear heating and enormous mechanical loads during plasma disruptions (EM-loads). The development of the structural design is accompanied by computational analyses, prototype manufacturing and prototype testing. Especially for structures with complex geometry, the optimum manufacturing route has to be determined regarding both technical and economical aspects.

On the basis of a characteristic section of the BSM (called BSM corner prototype) different manufacturing routes were evaluated. The front part of the BSM is exposed to volumetric nuclear heating of up to 3 W/cm^3 and to guarantee sufficient cooling it is designed with a double wall shell, which allows heat removal by internal meandering cooling circuits. Concerning the requirements on leak tightness, mechanical strength and geometrical precision, manufacturing of a double wall structure with internal cooling channels is a main challenge. Two BSM-corner prototypes were manufactured one by brazing of semi-finished parts and the other one by Hot Isostatic Pressing (HIP) and dedicated tests regarding pressure loss and cooling efficiency were performed. For the experimental part the Launcher Handling and Test Facility (LHT) at KIT was used to test prototypes under ITER relevant conditions. The results were compared with appropriate calculations and numerical FEM/CFD analyses performed with respect to flow characteristics, heat transfer and pressure loss of cooling water.

In this paper the experience obtained on prototyping is presented and the importance of prototyping for design validation is pointed out.

ITR/P1-13

The ITER Neutral Beam Test Facility in Padua – Italy: a Joint International Effort for the Development of the ITER Heating Neutral Beam Injector Prototype

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The development of the neutral beam injectors for ITER presents several challenging issues on physics and technology. Initially two Neutral Beam Injectors (NBI) systems will be installed on ITER, each designed to deliver 16.5 MW up to one hour to the ITER plasma. To fulfil this requirement the beam source has to provide a current of 40 A (46 A) of negative ions, D^- (H^-) and an electrostatic accelerator to 1 MeV (0.87 MeV).

These requirements have never been achieved simultaneously and therefore, in order to minimize the risks and the time to provide ITER with reliable NBIs, the necessity of a strong experimental demonstration and the necessity to optimize critical components and systems has been strongly endorsed by ITER and the ITER parties involved in the development of the neutral beam injectors.

A Neutral Beam Test Facility has been proposed, and this is now entering the construction and procurement phase, following the recent formal endorsement of the necessary Additional Direct Investment by the ITER Council. The Neutral Beam Test Facility will be hosted in Italy in the CNR research area of Padua where the Consorzio RFX operates.

The neutral beam test facility will host two experimental test stands, SPIDER (Source for the Production of Ions of Deuterium Extracted from an RF plasma) and MITICA (Megavolt ITer Injector and Concept Advancement).

The first test stand will optimize a full size negative ion source to full power and pulse length in H^- or D^-. Its mission is to demonstrate the capability to create and extract a current of D^- (H^-) up to approximately 50 A (60 A).

The second test stand is the prototype of the ITER injector, which aims at developing all the knowledge and technologies to guarantee the successful operation of the two injectors to be installed on ITER.

ITR/P1-14

Experimental Investigation And Validation of Neutral Beam Current Drive for ITER through ITPA Joint Experiments

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Joint experiments investigating the off-axis neutral beam current drive (NBCD) capability to be utilized for advanced operation scenario development in ITER was conducted in 5 tokamaks (ASDEX Upgrade (AUG), DIII-D, JT-60U, MAST (and NSTX)) through the international tokamak physics activity (ITPA). The following results were obtained in the joint experiments, where the toroidal field, B_t, covered $0.3 - 3.7$ T, the plasma current, I_p, $0.6 - 1.2$ MA, and the beam energy, E_b, $67 - 350$ keV. A current profile broadened by off-axis NBCD was observed in MAST. In DIII-D, good agreement between the measured and calculated NB driven current profile was observed. In JT-60U, agreement between measured and calculated NBCD location was obtained, when the NBCD location ($0.3-0.6$ in r/a), heating power ($6 - 13$ MW), triangularity ($0.25 - 0.45$), and E_b (85 and 350 keV) were widely scanned. In AUG (at low triangularity ~ 0.2) and DIII-D, introduction of a fast ion diffusion coefficient of $D_b \sim 0.3 - 0.5$ m^2/s in the calculation gave better agreement at high heating power (5 and 7.2 MW), suggesting anomalous transport of fast ions by turbulence. It was found through these ITPA joint experiments that NBCD related physics quantities reasonably agree with calculations (with $D_b = 0 - 0.5$ m^2/s) in all devices when there is no MHD activity except ELMs.

ITR/P1-15

Development of Full-size Mockup bushing for 1 MeV ITER NB System

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For verification of feasibility and a voltage holding capability of the high voltage bushing in the ITER NB system, manufacturing of a mockup and voltage holding test were performed at JAEA. A large bore ceramic ring (1.56 m in outer diameter) as an insulator of the high voltage bushing has been successfully brazed with 3 mm thick ring-shaped Kovar sleeves. After accomplishment of brazing of the ceramic, full-size mockup bushing was manufactured. Voltage holding of the mockup bushing at −200 kV for 3600 s has been demonstrated in the single-stage full-size mockup bushing. Vacuum insulation in a large insulator was discussed from the view point of electric field design toward 1 MeV ITER NB system.

ITR/P1-16

TSC Modelling of Major Disruption and VDE Events in NSTX and ASDEX-Upgrade and Predictions for ITER

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In the 2008 IAEA FEC, we had presented results of TSC simulations of fast MDs and slow VDEs and compared these simulation results with that obtained from DINA modelling. These results largely showed similar plasma behaviour, although somewhat differed in the predictions for the plasma current quench times and halo current magnitudes. Thus, it was decided to update both the models after benchmarking them with experimental observations in NSTX and ASDEX Upgrade (AUG) devices and use the updated codes to make more accurate predictions for ITER. Also ITER machine has undergone significant changes in the last year, e.g., in the vacuum vessel, blanket modules, central solenoid, divertor dome structure, addition of in-vessel control coils and so on, which affect the vertical evolution and disruption behaviour of the plasma. We present in this paper the TSC modelling of the VDE and MD events in NSTX and AUG devices, which help in improving and validating the models used in the code. The predictive modelling results for ITER with the updated TSC code, including the force predictions, are also presented.

ITR/P1-17

ITER Machine Assembly – Status and Plans

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Assembly of the ITER Machine aims at minimizing magnet error fields, ensuring alignment of the magnet system, and in-vessel components, ensuring compliance with the many code, QA and other technical requirements all within very demanding schedule and budget constraints.

As well as the basic machine, ITER consists of many associated plant systems installed within various buildings. To ensure safe working and allow optimisation of labour resources, tools and schedule overall responsibility for the coordination of all plant installation, and also the execution of installation tasks under IO responsibility, is centralised.

Whole lifecycle management of plant and components from installation to operation and maintenance is being implemented through Integrated Logistics Support, a strategy developed in military and large industrial projects. This strategy is fundamental in helping ensure the reliability of ITER and therefore demonstrate the feasibility of fusion.

To support assembly and installation there will be several hundreds of purpose-built tools, the designs of which are progressing. As well as the above interfaces, tool design is driven by component requirements, assembly sequences and constraints and the need to maintain the project schedule.

As well as these technical challenges, the assembly of the ITER defines the crucial latter part of the critical path of the project schedule. It therefore drives the start of Integrated Commissioning and ultimately the date of first plasma. Details will be given of how the assembly schedule has been challenged and modelled to identify near-critical path tasks which increase risk.

ITR/P1-18

Progress on the Development of the ITER Control System

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The development of the ITER control system has accelerated. A successful conceptual design review of the Control Data Access and Communication (CODAC) system was concluded in November, 2007, and confirmed the clearly defined three vertical tiers and two horizontal layers of the system. The vertical tiers distinguish the control, interlock and safety functions. The horizontal layers separate the central control and the plant system control. The central control is under the responsibility of IO, while most of the plant system control is under the responsibility of the seven ITER member parties. This responsibility sharing poses a major challenge for realization and integration of the

complete ITER instrumentation and control (I and C) system. In order to mitigate this risk, IO now enlarged the original scope of CODAC group by introducing the responsibility of supporting the plant systems to define their I and C specifications and help follow them through production and different acceptance tests and commissioning. Prerequisite to this activity is standardization in terms of hardware, software and methodology to be accepted by the ITER member parties. Initial progress has been made in this area of standardization.

The aim of this paper is to provide a status update of the ITER control system after inclusion of changes during the conceptual design review of 2007 and the increased functional capabilities.

ITR/P1-19

Development of the ITER Baseline Inductive Scenario

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Sustainment of $Q = 10$ (fusion power/input power) operation with a power level of $\sim$ 500 MW for several hundred seconds is a key mission goal of the ITER Project. Past calculations and simulations predict that these conditions can be produced in high-confinement mode operation (H-mode) at 15MA relying primarily on inductive current drive. Detailed predictive simulations, supported by experimental demonstrations in existing tokamaks, have been used to assemble an end-to-end specification of the scenario consistent with the final design of the ITER device. Simulations have encompassed plasma initiation, current ramp-up, plasma burn and current ramp-down, and have included density profiles and thermal transport models consistent with edge pedestal conditions present in current fusion experiments that maintain quasi-stationary conditions due to the presence of edge localized modes. Convergence on final designs for the ITER central solenoid and poloidal field coils has allowed the operating space for 15MA high-Q operation to be specified more accurately. To assess operational flexibility within the final operating space, several variations of the reference time-dependent scenario simulations have been analyzed to vary assumptions concerning pedestal parameters and to explore alternative operation on ITER. A range of pedestal transport assumptions was simulated in order to bound the range of operation due to uncertainties in the prediction of pedestal formation. The simulations also provide detailed information for final modifications of engineering subsystems that are being implemented on ITER. Data from these free-boundary control simulations yield configurational data required to assess first wall designs and diagnostic implementations. ITER will have an initial period of non-nuclear operation in hydrogen or helium to allow commissioning of all tokamak systems with plasma in advance of deuterium and deuterium-tritium operation. This paper will discuss details of these scenario simulations and the operational space boundary calculations for ITER operation. These conclusions will be illustrated by detailed comparisons of simulations that predict an operating envelope for ITER inductive discharge conditions.

ITR/P1-20

Current Ramps in Tokamaks: from Present Experiments to ITER Scenarios

Imbeaux, F.[1], Basiuk, V.[1], Budny, R.[2], Casper, T.[3], Citrin, J.[4], Fereira, J.[5], Garcia, J.[1], Fukuyama, A.[6], Hayashi, N.[7], Hobirk, J.[8], Hogeweij, D.[4], Honda, M.[7], Hutchinson, I.H.[9], Jackson, G.[10], Kessel, C.E.[2], Köchl, F.[11], Litaudon, X.[1], Lomas, P.J.[12], Lönnroth, J.[13], Luce, T.[10], Mikkelsen, D.[2], Miyamoto, S.[7], Nakamura, Y.[14], Nunes, I.[5], Parail, V.[12], Pereverzev, G.[8], Peysson, Y.[1], Polevoi, A.[3], Politzer, P.[10], Schneider, M.[1], Sips, G.[15], Tardini, G.[8], Voitsekhovitch, I.[12], Wolfe, S.M.[9], ASDEX Upgrade Team , C-Mod Team, DIII-D Team , JET–EFDA Contributors[16], JT-60U Team , Tore Supra Team , Contributors of the EU-ITM ITER Scenario Modelling Group , ITPA "Integrated Operation Scenarios" Group Members and Experts and ITPA , and "Transport and Confinement" Group Members and Experts

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In order to prepare adequate current ramp-up and ramp-down scenarios for ITER, present experiments from several tokamaks (ASDEX Upgrade, C-Mod, DIII-D, JET, JT-60U, Tore Supra) have been analysed by means of integrated modelling in view of determining relevant heat transport models for these operation phases. Two of the tested semi-empirical models are found to provide reasonable agreement relative to this criterion, i.e. are able to model the experimental dataset within ± 0.15 agreement on $l_i(3)$. The Coppi-Tang-Redi model is found to be less accurate and provides more deviation on $l_i(3)$. Conversely, first-principle based models such as GLF23 are found to be difficult to use at very low current and close to the plasma edge, resulting in strong deviations in the internal inductance, this one being strongly weighted by the current density close to the edge. By patching the model in the region $\rho = 0.8 - 1$, one can however use it for prediction with an accuracy on $l_i(3)$ similar to that of the empirical models. Finally, projections to the ITER current ramp-up and ramp-down phases are carried out with the most successful models. The ITER current ramp-up simulations include ohmic as well as L-mode cases with various heating schemes, in particular Electron Cyclotron Resonance Heating and Lower Hybrid Current Drive. Though significant differences between models appear on the electron temperature prediction (in particular inside the ECRH deposition), the

final q-profiles reached in the simulation are rather close. The difference between models on the $l_i(3)$ prediction is also small, of the same order as for the present experiments, i.e. ± 0.15. Finally, the sensitivity of the projections on various parameters such as Z_{eff}, density profile peaking, initial equilibrium and boundary conditions is quantified.

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ITR/P1-21

Progress in Development of the Blanket Remote Handling System for ITER

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R&D to clarify the specifications of the in-vessel transporter (IVT) design in detail has been performed with feasible outputs, i.e., the demonstration of final installation to within 0.5 mm between the blanket module and two keys by using torque control to avoid any jamming as a result of excessive loading during the module installation process. In addition, rail connection and cable handling in the transfer cask, which are critical issues for IVT system, have been demonstrated from view points of the accurate connection of rail joint and compact design for cable handling mechanism to handle the cable of 75 mm in diameter.

ITR/P1-22

Development of ITER Advanced Hybrid and Steady State Scenarios

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ITER will provide critical information on advanced burning plasma scenarios of interest for fusion power plants by pursuing hybrid and steady state discharges. Full discharge simulations are performed to examine the plasma current rampup, flattop and rampdown phases self-consistently with the poloidal field (PF) coils and their limitations, plasma transport evolution, and heating/current drive (H/CD) sources. These time dependent self-consistent free-boundary calculations are performed with the Tokamak Simulation Code (TSC) and PTRANSP transport evolution codes coupled together. The steady state scenario in ITER targets a plasma current around $8-9$ MA, and is found to require energy confinement multipliers relative to IPB98$_{(y,2)}$ in the range of 1.4 to 1.8, depending on the pedestal and core transport model. The plasma density needs to be in the vicinity of the Greenwald density limit in order to provide sufficient fusion power, with

Q becoming less than 3 when n/n_{Gr} drops to 0.8. Providing 100% of the plasma current non-inductively requires upgrades to the day-one heating and current drive complement (up to 35 MW of lower hybrid and/or 20 additional MW of EC are considered). The pedestal temperature range is $1.5-4.0$ keV, and the density profiles range from very flat to slightly peaked with $n_0/\langle n\rangle = 1.05-1.5$. The hybrid scenario in ITER is basically an ELMy H-mode, and targets a plasma current in the range of $12-13$ MA, while obtaining an energy confinement multiplier relative to IPB98$_{(y,2)}$ scaling of $1.0-1.25$, depending on the pedestal and core transport model (GLF23 or Multi-Mode 08). The hybrid scenario may allow a high fusion power without operating at the highest plasma current. The effects of plasma rotation, impurities, and a range of T_i/T_e are examined in the transport models. The temperature pedestal is examined in the range of 2.1 to 5.7 keV. The day-one heating and current drive sources (NB, EC and ICRF) are used for this scenario, providing $35-60\%$ of the plasma current non-inductively. Other techniques to access the $q_0 \sim 1$ regime will be examined, such as sawtooth destabilization and off-axis current drive. The primary task is to identify time-dependent full discharge operating scenarios and the equilibrium operating space available for these hybrid and steady state plasma discharges in ITER within the allowable limits.

ITR/P1-23

Progress in High Energy Load Test of Beryllium, Tungsten and CFC for ITER First Wall and Divertor

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Analysis and extrapolation of measurements in present tokamaks have been used to derive the heat load specifications on ITER plasma facing components for which the expected static heat fluxes parallel to the field on the first wall panels can reach values of up to 8 MWm^{-2}, while the ELM averaged parallel heat flux reaches values of up to 25 MWm^{-2}. On the other hand, transient power fluxes on to the divertor targets by ELMs are expected to be ~ 0.5 MJm^{-2} for controlled ELMs and ~ 20 MJm^{-2} for uncontrolled ELMs. The present paper describes qualification tests for First Wall and Divertor panels performed both in e-beam facilities (for stationary loads) and in plasma gun facilities (for ELM and disruption loads). In accordance with Procurement Arrangement, RFDA will start in

2010 High Heat Flux tests of divertor elements – Dome divertor, Outer and Inner vertical target on the new e-beam facility – IDTF (ITER Divertor Test Facility). The power of the e-beam is $\sim$ 800 kW, and cyclic heat load of more than 20 MWm^{-2} will be applied onto the divertor components. For the first wall panels, Russian type of beryllium TGP-56 FW was tested in e-beam facility JUDITH (Jülich) as a possible ITER material alternative to S65C VHP beryllium. The results of qualification TGP-56 FW demonstrated reliable performance capabilities as plasma-facing material for the ITER First Wall. To obtain data for the empirical evaluation of the beryllium PFC lifetime and dust production under ITER-relevant transient loads, a new plasma gun facility (QSPA-Be) has been put into operation. The results of experiments with Be-coated tungsten and Be-clad components at the QSPA-Be facility will be presented in the paper and compared with those for CFC and W obtained earlier at QSPA-T facility. Additional important tasks, which can will be addressed with this new facility are the study of tritium retention in Be-clad components its erosion products as well as the development of methods for removing tritium from them.

ITR/P1-24

Consequences of Deuterium Retention and Release from Be-containing Mixed Materials for ITER Tritium Inventory Control

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Control of in-vessel tritium retention is one of the critical issues for ITER. Since the current design of ITER plasma facing components includes the use of different materials hydrogen retention will be strongly affected by formation of mixed material deposits. In ITER, wall baking at 240°C for the main chamber and 350°C for the divertor is suggested as a possible tritium removal method. To assess the efficiency of such wall baking procedures, the tritium release behaviour from Be-containing mixed materials was investigated under controlled laboratory conditions.

Be films on W/graphite/CFC substrates were deposited with a thickness of several hundred nanometers. Be-containing compound layers (Be12W and Be2C) were subsequently created by thermal annealing of Be-W and Be-C samples. Low energy D ion (200 eV) implantation of the samples was performed to fluences up to 5×10^{23} D/m^2. The sample temperature was kept fixed during implantation at chosen values from RT to 350°C. D retention and its release behaviour was investigated by means of thermal desorption spectroscopy (TDS).

At 350°C, the released fractions from pure Be films implanted at room temperature are typically $> 90\%$ of the total retention showing a strong release peak already at temperatures of $150 - 200$°C. Even at a temperature of 240°C, as expected at the vessel walls, the retained amount is below 20%. The fraction retained at 350°C annealing increases to $> 50\%$ for implantation temperatures above 150°C.

The fraction of D released by baking to 350°C is strongly influenced by intermixing and compound formation of the Be layer with the substrate material. In the case of compound layers the retained fraction during TDS increases. Both, for the formation of Be2C and Be12W only about 50% of D is released at 350°C. For Be2C two different trapping regimes are evident: a strong release at 175°C, equivalent to pure Be, and a second component with higher trapping energy retaining deuterium even at temperatures above 500°C, equivalent to pure C. The release from Be12W, a compound that could potentially form on the W divertor plates in the activated phase of ITER, proceeds similar to the release from pure W.

These results and detailed parameter studies as function of fluence and their consequences for the tritium removal during the wall baking procedure in ITER will be discussed in the presentation.

ITR/P1-25

ECRH Assisted Plasma Start-up with Toroidally Inclined Launch: Multi-Machine Comparison and Perspectives for ITER

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ECRH assisted plasma break down is foreseen with full and half magnetic field in ITER. Therefore, the corresponding O1- and X2-schemes have been studied in present day devices with respect to pre-ionisation and breakdown assist. Earlier experiments, presented at the 2008 IAEA FEC, proved the feasability in principle. This contribution reports on an extension of these common experiment with regard to the toroidal inclination of the ECR beam, which is 20 degrees in ITER. All devices could demonstrate successful breakdown assist also for this case, although the necessary power was up to 100% higher compared to perpendicular launch. As the experiments were executed, differences between the devices with regard to the required power and vertical field became obvious, which may also be important for ITER. Since results of the single devices are in parts already published, the main emphasis of this contribution will we put on their comparison, common conclusions and unresolved differences.

ITR/P1-26

Disruption, Halo Current and Rapid Shutdown Database Activities for ITER

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Disruption characterization and database development and analysis activities conducted for ITER under the aegis of the International Tokamak Physics Activity (ITPA) Topical Group on MHD Stability are described. An ITPA International Disruption Database (IDDB) Working Group and a MDSplus-based IDDB infrastructure for collection and retrieval of disruption-relevant tokamak data, first established in 2006, comprises one of the several "joint experiments" being conducted by the MHD Topical Group. Analysis reported in 2006 of the current quench data from 7 elongated-plasma tokamaks provided a new recommendation about the lower bound on the plasma current decay time expected in ITER. In 2010, expansion of the scope and content of the IDDB to encompass halo current data has been initiated, and new combined current decay and halo current data sets (shot records) from the seven previously-contributing tokamaks are expected. The analysis to be reported in this paper will provide an "integrated" current decay and halo current basis for updated recommendations to the ITER Organization in regard to ITER halo current magnitude and toroidal asymmetry and how these attributes correlate with the parent plasma aspect ratio, elongation and triangularity and current and toroidal field magnitudes. The feasibility of interpreting database composite and device-specific data in terms of a "statistical" load severity spectrum will be explored. Activity has been

initiated to add IDDB data categories for rapid plasma shutdowns effected by massive gas and pellet injection.

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ITR/P1-27

Shattered Pellet Disruption Mitigation Technology Development for ITER

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Disruptions on ITER present challenges to handle the intense heat flux, the large forces from halo currents, and the potential first wall damage from multi-MeV runaway electrons. Injecting large quantities of material into the plasma during the disruption can reduce the plasma energy and increase its resistivity and electron density to mitigate these effects. Developing the technology to inject sufficient material deeply into the plasma for a rapid shutdown and runaway electron collisional suppression is an important capability needed for maintaining successful operation of ITER. A shattered pellet injection technology is being developed at ORNL that can be employed on ITER as part of a system to provide mitigation of disruptions. Here we present progress on the development of a shattered pellet injection system that produces large solid cryogenic pellets to provide reliable deep penetration of material for disruption mitigation. A concept for the application of this tool on ITER is discussed.

The shattered pellet injection technique utilizes a pipe-gun type injector that forms a large cryogenic pellet in-situ in the barrel. The pellet is accelerated by a high pressure gas burst and hits metal plates that are optimized to produce a spray of solid fragments mixed with gas and liquid at speeds approaching the sound speed of the material. The divergence and particle distribution of the plume of shattered material has been measured and optimized. The shattered cryogenic pellet concept has recently been shown on DIII-D [1] to lead to deep penetration and high density that shows promise for rapid shutdown and disruption mitigation in ITER.

A conceptual system of this type for ITER with multiple barrels is under investigation for use on ITER to mitigate disruptions and suppress the formation of runaway electrons. Such an injector with multiple barrels is small enough to easily fit inside an ITER port plug for close coupling to the plasma, which is desirable for faster mitigation. Integration issues with other ITER subsystems and possible locations on the machine are presented.

[1] N. Commaux, et al., submitted for publication.

ITR/P1-28

ITER Fuelling System Design and Challenges – Gas and Pellet Injection and Disruption Mitigation

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The ITER fuelling system plays a key role in plasma operation, ensuring density control, ELM frequency control, radiative cooling enhancement, plasma detachment control, disruption mitigation, etc. It consists of 3 major sub-systems: the Gas Injection System (GIS), Pellet Injection System (PIS) and Disruption Mitigation System (DMS). This paper describes the design status of the three systems and the challenges associated with each. The ITER fuelling system is capable of delivering fuel particles (H_2, D_2 and DT) at average and peak throughputs of 200 Pa·m^{3}s^{-1} and 400 Pa·m^{3}s^{-1} respectively in the form of gas or pellets, as well as impurities such Ne, Ar and N_2 with average and peak throughputs of 10 Pa·m^3 s^{-1} and 100 Pa·m^3 s^{-1}.

The current GIS consists of 4 upper port level and 3 divertor port level injections. During burning plasma operation, boundary plasma simulations suggest that gas fuelling from the edge will be inefficient for core fuelling, even for main chamber injection locations where fuelling efficiencies are usually higher. The upper port injections are thus foreseen as vehicles for possible He ash removal (increasing the SOL density independently of the core) or for coupling improvement of RF heating systems. The divertor injection points are envisaged mainly for extrinsic seeding of impurities to effect detachment control through volumetric radiative cooling.

The PIS provides core plasma density control from high field side (HFS) injection and ELM pacing from low field side (LFS) pellet introduction. The current flight tube configuration for HFS pellet injection allows maximum pellet speeds of 300 ms^{-1} to be achieved.

The very high plasma stored energies of which ITER will be capable mean that mitigation of thermal and electromagnetic loads due to disruptions, vertical displacement events and runaway electrons is indispensable for machine protection. Physics studies to define the requirements for the DMS are currently running in parallel with a detailed engineering assessment of candidate systems.

ITR/P1-29

Benchmarking ICRF Simulations for ITER

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Benchmarking simulations of ICRF in ITER plasmas are performed using plasma profiles and equilibria obtained from integrated self-consistent modeling predictions of two ITER H-mode plasmas. One is for a high performance baseline (5.3 T, 15 MA) DT plasma, and the other is for a half-field, half-current H-only plasma of interest for the pre-activation phase. The heating scheme uses He^3 at $2-3\%$ and 52 MHz. The predicted profiles are used independently by six groups to simulate the ICRF electromagnetic fields and plasma heating profiles. Approximate agreement is achieved for the predicted heating power partitions. The partitions for heating impurities and fast ion species are small according to the simulations. Comparisons of heating profiles are given. Consequences of altering the minority fraction are discussed. The requirements, for accurate simulations, of numerical grid resolution and of including realistic geometry and plasma ion species are discussed.

ITR/P1-30

ELM Control by Resonant Magnetic Perturbations: Overview of Research by the ITPA Pedestal and Edge Physics Group

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This paper presents a multi-tokamak overview of experimental results and planned hardware upgrades that will provide the essential physics understanding needed to project resonant magnetic perturbation (RMP) ELM control to ITER. Reduction of ELM size by at least a factor of 20 is critical to achieve acceptably low erosion of ITER material surfaces. The joint work reported here is part of the plan formulated by the PEP ITPA and ITER IO to provide the physics basis for the proposed use of internal RMP coils for ITER ELM control. This work includes three critical elements, (1) reproduce ELM suppression with RMP fields in at least one tokamak other than DIII-D, (2) identify the criteria for ELM suppression from both experimental results and theory, and (3) assess the impact of the use of RMP fields for ELM control on a) pedestal pressure and core confinement, b) divertor and main chamber wall loading, c) ELMs during the I_p ramp and/or near the L-H power threshold, and d) core pellet fueling. Modifications of ELMs with RMPs were achieved in JFT-2M, DIII-D, JET, MAST, and NSTX including (i) suppression of ELM energy losses with internal coils in DIII-D, (ii) mitigation of ELM size with external coils in JET and with internal coils in both MAST and DIII-D, and (iii) pacing of ELMs with modulated RMP pulses in NSTX and DIII-D. The experiments in MAST and JET confirm that the island overlap width condition, correlated with ELM suppression in DIII-D and used to guide the requirements of ITER RMP coils, is not sufficient to assure ELM suppression in multiple devices. Progress in theory indicates that 3D field structure including the resonant plasma response may be important. For 2011, upgrades of the internal RMP coil systems are underway on DIII-D, MAST and AUG that will permit greater variation of RMP mode spectrum to test ELM control physics models. Three rows will be installed on the centerpost of DIII-D, the MAST coils will be upgraded above the outer midplane, and AUG will use a staged installation of coil rows above, below and on the outer midplane similar to the ITER design. These systems will greatly increase the capability to test theoretical models of RMP ELM control and the probability of achieving ELM suppression on multiple tokamaks.

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ITR/P1-31

Experimental Simulation of ITER ELMs Impacts to the Tungsten Surfaces with QSPA Kh-50

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In this report, recent results of ELM-simulation experiments with QSPA Kh-50, largest and most powerful device of this kind, are presented. Crack patterns (major- and micro-) in tungsten targets and cracking thresholds (both threshold energy load for cracking onset and threshold target temperature related to DBTT) as well as residual stresses after repetitive plasma pulses have been studied for a deformed W material (Plansee AG). Monolithic tungsten which is manufactured according to this technology is considered as ITER-reference grade. The elongated grain orientation was perpendicular to the surface. The targets were preheated at 200°C, 300°C, 400°C or 600°C, to estimate effects of Ductile-to-Brittle Transition Temperature (DBTT) on cracking failure. The tungsten surface was analyzed after 1 pulse, 5 pulses and 10 pulses of $t = 0.25$ ms time duration to investigate the influence of material modification by initial plasma exposures on cracking thresholds. The magnitudes of heat load were 0.2 MJ/m^2, 0.33 MJ/m^2, 0.45 MJ/m^2 (which did not cause melting) and 0.75 MJ/m^2 (above the melting threshold).The thickness of major and micro cracks, the network distance as well as cracks penetration to the material depth are analyzed. Comparisons of cracking failure of deformed tungsten with behaviour of sintered W samples are performed. Cracking development is characterized by measured threshold load and threshold target temperature, which determine the existing region of W performance without cracks. In these terms, deformed W material is found to be more resistive against cracking and grain losses as compare to other grades. Residual stresses do not practically depend on initial target temperature but essentially grow with increasing thermal loads, and cracks are not developed if the residual stress in result of plasma pulse is below $300 - 350$ MPa. The obtained experimental results are used for validation of the numerical code PEGASUS 3D.

ITR/P1-32

Characterization of Runaway Electrons in ITER

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Numerical simulation of Runaway Electrons (RE) produced during plasma disruption in ITER has been carried out with the aim to develop means for their suppression and/or mitigation of the potential damage of Plasma Facing Components (PFC). The simple models for RE generation used previously [1] have been upgraded to include Monte-Carlo kinetic treatment of RE tails in 2D geometry of plasma equilibrium, OFMC modules for analysis of the heat loads on 3D first wall and divertor targets, MHD stability analysis of RE discharges. The new model has better description of nuclear source of RE due to Compton scattering of gammas and tritium decay as well as initial Dreicer acceleration of the tails and scattering of relict energetic tails that could survive thermal quench of disruption. Model has been validated on experimental results from the present tokamaks.

It has been confirmed that avalanche of runaway electrons results in generation of massive RE current in ITER even during disruptions with $I_p = 9$ MA and lower. Analysis of MHD stability of RE current during plasma VDE allowed evaluate fraction of magnetic energy that could be transformed in the kinetic energy of RE electrons during disruption and VDE. Energy and pitch angle spectra calculated by kinetic codes together with evaluation of the most probable poloidal distribution of the RE's on the PFC provides a background for future evaluation of potential damage of the first wall tiles during unmitigated disruption.

Different means for suppression of RE's or mitigation of their impact on the first wall are discussed. Simulation of plasma equilibrium control of RE discharge has been carried out with the aim to investigate controllability of RE discharges after major disruptions. Modelling shows that ITER system for vertical stabilization is capable to control only limited range of RE current profiles. Simulation of RE suppression by repetitive injection of dense gas jets in the current quench phase has been carried out and provided requirements for range of jet parameters and their repetition rate.

[1] V. Lukash, Report, ITER-D-2LPK4Yv1.0, 2000.

ITR/P1-33

ITER Divertor Performance in the Low Activation Phase

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A low activation phase of ITER operation is an important part of the ITER research plan in order to commission and test ITER sub-systems such as those for ELM mitigation and additional heating. During this phase, deuterium or tritium fuelling is not permitted and no fusion power is thus available. Operation in H mode (to produce ELMs) is to be facilitated by the use of helium plasma at half current and half field. In these He plasmas, ELM pace-making by pellet injection can only be tested by adding core fuelling by H pellets to the edge fuelling of He from gas puffing. To investigate the operating window accessible to He plasmas with an H admixture, 2D divertor simulations and consistent 1D core simulations have been performed. It is found that the peak target power loading can reach the values expected for the full-power D-T case, so that a realistic assessment of the divertor power handling capacity relevant for later operation can be performed very early on. A variation of the H content in the mostly He plasma shows a rather weak effect of the H concentration on this parameter. The roll-over of the ion saturation current on the targets (an indication of partial detachment) at the highest H concentration occurs at a neutral pressure similar to that for DT, but shifts to higher pressure for almost pure He. The maximum temperature at the targets is higher with He than with DT for the same neutral pressure. One of the principal questions to be answered by modelling is the strength of core fuelling, which determines the density control in the discharge. Since this also depends on the core transport, the modelling work necessarily involves development and application of a consistent core-edge model for the low-activation phase plasmas. We use the parameterized results of the SOLPS modelling as the boundary conditions for the 1D core-and-pedestal model. First results indicate that He fuelling via gas puffing will be insufficient under some conditions to maintain the required density in the core in the absence of a strong inward pinch, which might occur in the experiment but for which no validated physics basis yet exists. In this case, hydrogen pellets could be used to maintain the necessary electron density in the He-H plasma. Compatibility of the dilution of the He plasma with maintenance of a good quality H-mode in He plasma is assessed.

ITR/P1-34

Electron Cyclotron Power Losses in ITER for 2D Profile of Magnetic Field

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Recent comparison [1] of numeric codes SNECTR, CYTRAN, CYNEQ and EXACTEC for calculating the 1D distribution, over magnetic flux surfaces, of the net electron cyclotron (EC) radiated power density, $P_{ec}(\rho)$, was carried out for a flat 1D profile of total magnetic field, which is an average over each magnetic surface and is used in 1.5D transport models: $B_{tot}(\rho) = B_T(Ro) = B_o$, vacuum toroidal magnetic field on toroid's axis. However, the predicted rise of T_e in steady-state operation regimes in ITER and future reactors requires better accuracy of $P_{ec}(\rho)$ calculations, especially in the plasma hot core. Here we take into account the inhomogeneity of magnetic field in 2D approximation in the modified code CYNEQ. We compare three approximations of the magnetic field profile: (i) 2D - magnetic field $B(R, Z)$; (ii) 1D – flux surface averaged magnetic field $B(\rho) = \langle B(R, Z)\rangle$; (iii) 0D – homogeneous magnetic field $B = B_o$. It is shown that, for the same plasma parameters expected in ITER, the $P_{ec}(\rho)$ profiles in the cases (i) and (ii) are very close, being lower than $P_{ec}(\rho)$ in the central plasma in the case of $B = B_o = 5.3$ T. In particular, for the enhanced confinement scenario [2], the decrease reaches $\sim 25\%$. The above effect has to be taken into account in the 1.5D transport codes when modeling the steady-state regimes of ITER operation. When central temperature increases to ~ 30 keV the local EC power loss becomes a substantial part of heating from fusion alphas and exceeds the auxiliary heating from neutral beam. On the other hand, fast increase of $P_{ec}(0)$ with temperature has a positive impact on stabilization of fusion burning. Thus, for reactor scale parameters, accurate simulations of nonlocal heat transport by EC waves requires self-consistent 1.5D calculations of plasma parameters with 2D equilibrium. Meanwhile the EC energy transport has sufficient accuracy with 1D surface-averaged magnetic field $B(\rho) = \langle B(R, Z)\rangle$, derived from self-consistent 1.5D simulations. It helps noticeably reduce computational time in the time dependent 1.5D simulations of plasma scenarios in comparison with full 2D EC transport simulations keeping the required accuracy.

[1] Fusion Sci. & Technol., 2009, v.55, p.76

[2] 19th IAEA Conf. on Fusion Energy, 2002, CT/P-08

ITR/P1-35

Integrated Modeling of Steady-State Scenarios and Heating and Current Drive Mixes for ITER

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Recent progress on ITER steady-state scenario modeling by the International Tokamak Physics Activity (ITPA)/Integrated Operation Scenario (IOS) Topical Group is reviewed. One of the primary goals of the ITER project is to demonstrate reactor scale steady state (SS) operation for future tokamak reactors, specifically long pulse ($\geq$ 3000 s) operation with full noninductive current, high fraction of bootstrap current, and fusion gain $Q \geq 5$. We present integrated modeling of such steady-state scenarios for ITER parameters, using various theory-based and semi-empirical transport models in conjunction with models for the heating and noninductive current drive sources available to ITER. The modeling is validated by comparing the simulations with the present-day experiments with similarly high noninductive operation at high beta. A new weak magnetic shear scenario has been derived by an efficient steady state solution method using the theory-based transport model GLF23 with scaled experimental boundary profiles from a self similar tokamak discharge. A second new scenario with strongly reversed magnetic shear was derived using a prescribed transport model that is validated against an experimental discharge. Benchmarking was carried out using the transport evolution codes ONETWO, CRONOS, PTRANSP, TASK, TOPICS, and ASTRA. Finally, the steady state solution method was exploited to examine steady-state scenarios for upgrade choices for the heating and current drive mix for satisfying the steady state objective.

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ITR/P1-36

3D Effect of Ferromagnetic Materials on Alpha Particle Power Loads on First Wall Structures and Equilibrium on ITER

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The finite number and limited toroidal extent of the TF coils cause a periodic variation of the toroidal field called the magnetic ripple. This ripple can provide a significant channel for fast particle leakage, leading to very localized fast particle loads on the walls. Ferromagnetic inserts will be embedded in the double wall structure of the vacuum vessel in order to reduce the ripple. In ITER the toroidal field deviations are locally further enhanced by the presence of discrete ferromagnetic structures, e.g. TBM. Thus, there are complex symmetry-breaking effects. It is not yet fully understood how superimposing the periodic ripple and a local perturbation affect the fast ion confinement and concerns have been voiced that the combined effect might lead to significant channelling of the alpha power. In this work, the wall power loads due to fusion-born alpha particles were re-studied for a variety of cases addressing issues such as different wall configurations, proper inclusion of the TBM effect on the magnetic background, and the possible corrections to 3D equilibrium introduced by the ferromagnetic materials using the 3D equilibrium code, VMEC, since 3D corrections to the equilibrium might enhance the alpha particle loss. To properly include the TBM effect on the magnetic background, the FEMAG code was used, and the effect was calculated on the total field including the poloidal field by the plasma current as well as the vacuum field. In the VMEC analysis, it was found that the difference between a full 3D equilibrium reconstruction and "an axisymmetric equilibrium + vacuum fields" was small. Thus, it was concluded that no 3D equilibrium reconstruction was needed and that it was sufficient to add the vacuum field perturbations onto an axisymmetric equilibrium. Under the new boundary condition, the wall load calculation was carried out by using ASCOT, DELTA5D, and F3D OFMC code. Including the plasma current contribution in the magnetic field calculation did not produce significant changes in the results. The peak heat load is < 50 kW/m^2, is well below the critical level of ~ 1 MW/m^2. Sensitivity of the peak heat load on the first wall shape was also observed. These results indicate the importance of our continuing contribution to the re-evaluation of power loads following updates in wall shape and equilibrium.

ITR/P1-37

Physics and Modeling of the Negative Ion Source fot the ITER Neutral Beam Injection

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This paper describes the work performed at LAPLACE on the modelling of the Negative Ion Source for the ITER neutral beam system. The aim is to develop a complete model of the source, including driver, diffusion chamber, magnetic filter and extraction, in order to better understand the physics of the source, contribute to its optimisation, and help in the interpretation of the experimental results.

ITR/P1-38

Results of the Prototype EB-Welded Segment for the ITER Vacuum Vessel

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The primary functions of the ITER Vacuum Vessel are to provide a high quality vacuum for the plasma as well as the first confinement barrier of radioactive materials. The Vacuum Vessel is a torus-shaped divided toroidally into nine 40° sectors and each sector is made of 4 segments named, Inboard, Upper, Equatorial and Lower.

With the purpose to confirm the manufacturing parameters and consequently to start in time the Vacuum Vessel sectors production a prototype EB-welded segment contract was placed by "European Fusion Development Agreement" in 2006.

This prototype EB-welded segment called "Vessel Advanced Technology Segment" (VATS) is a 316 LN IG trial version of the final Inboard segment of 6 meters long by 2 meters wide, 60 mm thick with a total weight of 17 tons.

The main aspects investigated by the VATS manufacturing were, the EB welding qualification procedures, the manufacturing method validation and the computer analysis calibration of the welding distorsions in comparison with the measured ones.

Highlights on the methodology take off to determine the optimum EB-welding sequence with a minmum of jigs based on a calibration heat source study developed on a smaller "Validation E-beam Welded Coupon" (VEC) are given.

In order to be able to achieve the required as-welded tight tolerances, this study has the principal goal to develop two computational numerical techniques using SYSWELD and ANSYS codes. Nine different simulation sequences were carried out to explain the

different mechanisms that drive the distortions process during welding and to lead to an optimum sequence that minimizes the final distortions.

Based on this practical and theoretical work, were can now generate and use to investigate many welding sequences with effect to optimise the Vacuum Vessel Sectors production and achieve the required tolerances.

ITR/P1-39

Unit Operation Analysis of the Tritium Plant Storage and Delivery System in ITER

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The storage and delivery system of the tritium plant in the ITER has major functions to handle the fuel gases, mainly tritium-contaminated gases, safely. The new strategy of fuelling gases and the regulation of construction site in the ITER have led the change of the process design of the storage and delivery system. There are three kinds of needs to change the process design of the storage and delivery system. Regarding the pressure of manifold from the storage and delivery system to the fuelling system, it was proposed that the pressure of fuel gases will be reduced from 1.2 bar to sub-atmospheric pressure to prevent the tritium release into gallery in case of the guillotine brake of the double contained fuelling manifold. Regarding the need of fuel gases in the fuelling system, DT is not needed any more and the new fuelling rate of each gas has been evaluated. Regarding the safety to use of tritium-contaminated gases, the strict inventory limitation will be applied to prevent the propagation of fire and to reduce of the release amount of the tritium-contaminated gases. This study focused on the process analysis to satisfy the change of the environments in the process design of the storage and delivery system: how to perform the required fuelling rates and conditions of fuels and how to satisfy the safety regulation and guidelines to treat tritium-contaminated gases.

ITR/P1-40

Study on the Impact of Plasma Disruption on the Current Control of the ITER Coil Power Supply

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In order to provide the current profile required for the ITER operation, the ITER coil power supply has to keep the continuous current during current inversion. There is potential risk of over voltage arising by the plasma disruption, which will destroy the converters in case the converter fails to keep the suitable conducting path. This paper investigates the impact of plasma disruption on the current control of the ITER coil power supplies for the MC converters. The disruption induced coil voltage for ITER CS and PF coils was reviewed and the induced voltage of a PF coil was implemented in the PSIM circuit simulation with the voltage source by analog behavior model. The dependency of the current gap between the converters in parallel was examined to find the limitation of the present current controller.

ITR/P1-41

Status of Design and R&D Activities for ITER Thermal Shield

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ITER Thermal Shield (TS) system plays the role of reducing the heat load transferred by thermal radiation and conduction from warm components to the magnet structures that operates at 4.5K. The TS consists of equatorial TS (ETS), upper TS and lower TS. The ETS has three sub-parts such as vacuum vessel TS (VVTS), equatorial cryostat TS (ECTS) and support TS (STS). Thermal radiation to the magnet structures is minimized by operating the TS at low temperature and by providing the TS surfaces with low emissivity using silver coating. ITER TS is planned to be procured completely by Korea. The Korea Domestic Agency (KODA) has been studied the TS design with the collaboration of industries since 2007. The preliminary design was completed in March 2010. Overview of preliminary design, supporting analysis and R&D statuses are described in this paper.

ITR/P1-42

Fabrication Design Progress of ITER Vacuum Vessel in Korea

Kim, B.C.[1], Sa, J.W.[1], Kim, H.S.[1], Lee, Y.J.[1], Hong, K.H.[1], Ahn, H.J.[1], Bak, J.S.[1], Jung, K.J.[1], Park, K.H.[2], Kim, T.S.[2], Lee, J.S.[2], Sung, H.J.[2], Kim, Y.K.[2], Kwon, T.H.[2], Ham, J.K.[2], Hong, Y.S.[2], Shin, S.B.[2], Kim, H.K.[2], and Kwon, I.K.[2]

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Korea as the in kind provider of the two ITER vacuum vessel (VV) main vessel sectors, equatorial and lower ports structures made the first procurement arrangement (PA) on VV. After the contract with Hyundai Heavy Industry for the VV fabrication, HHI with KODA do some R&D to develop the fabrication procedures. The two types of mock up to optimize the electron beam welding techniques will be fabricated. In parallel, the fabrication design is also undergoing with the special attention on the control of welding deformation. To meet the special requirement on the material, the supporting R&D with KO mill maker was also done.

ITR/P1-43

Current Activities on Design and Development of ITER Blanket Shield Block

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The conceptual design of blanket shield block has been completed in the framework of BIPT (Blanket Integrated Product Team) since the redesign of blanket system was recommended by 2007 ITER design review. As a part of blanket conceptual design tasks KO DA (Domestic Agency) has participated in the blanket electromagnetic benchmarking activity, coordinated by ITER organization, to verify the analysis methodology and to compare the calculation results. In addition the thermo-hydraulic and thermo-mechanical analyses of the in-board conceptual model were carried out to ensure the thermal performance. Regarding manufacturing technologies, fundamental fabrication techniques such as drilling, machining and deep electron beam welding were successfully developed in the close collaboration with industries. Based on the developed techniques a quarter mock-up was manufactured and fabrication feasibilities of the blanket shield block were also identified. This paper presents recent activities and progresses on the design and fabrication of the ITER blanket shield block in Korea.

ITR/P1-44

Status on the Development of the Fabrication Technology and the Mock-up Qualification Tests for the ITER Blanket First Wall

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ITER blanket modules are composed of first walls and shield modules. FWs are one of the demanding components in ITER, since they would receive the severe localized and cyclic heat loads during the operation. Therefore, the qualification program was agreed to be performed in order to guarantee the acceptable quality of the total FWs procured by six different domestic agencies including Korea. The manufacturing specification of first wall mock-up including design and the high heat flux testing plan were made by the discussion between IO and domestic agencies. Each first wall mock-up consists of three beryllium armor tiles attached to a Cu alloy (CuCrZr) heat sink plate internally cooled by SS cooling tube. The two mock-ups were delivered by Korea domestic agency to the test facilities (US and EU). Both of them were successfully passed in the qualification tests and their test results were approved by ITER Organization. In this paper, the status of the development of the fabrication technology and the results of the qualification tests for KO mock-ups were reviewed.

ITR/P1-45

R&D of the Fabrication Technology for ITER Magnet Supports

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The R&D of manufacture technology for ITER magnet supports is an essential step for construction. 316LN with high nitrogen content is recommended for magnet supports. China has developed both hot-rolled thick plate and large-size forged block. The material showed high strength and toughness at room temperature, 77K and 4.2K. In addition, almost no ferritic phase could be seen. A new design for GS manufacture without welding was carried out. The static stress analysis using three-dimensional FEM was carried out. The result shows that the stress in the present load condition/combination is under the stress limitation of the material. The flexible plates have no severe stress concentration under the normal operation, and even during accident. the bulking analysis shows that the structure is safe enough. However, engineering test of the stability of the prototype GS pedestal during various load combinations is still essential. For cooling pipe, it is suggested to use brazing connection to attach the cooling pipe to support-plates. We developed several kinds brazing filler as candidates, including Sn-Pb, Ag- based and

Cu-based alloy. The composition, microstructure, wet-angle, boundary condition after brazing, brazing strength at both room temperature and low temperature, and toughness were tested. The tensile strength of brazed parts is around 700 MPa at 77K for Ag-based and Cu-based fillers. In PF support system, PF3 and PF4 share the same support and suspended by a strut dowel to the TF coil case. The dead weight of PF coil combined with electromagnet force during discharge may enhance the movement of strut dowel. Ion implantation was utilized to modify the surface condition of the strut dowel due to its non-boundary modification of the microstructure as well as the formation of hard alloys on the surface. It is clear that the wear resistance was improved obviously after ion implantation vie increase the surface hardness and reduce the wearing. However, the test of the wearing resistance with the enhanced load at low temperature ($4 - 77$K) is still a challenge in present. In addition, the TIG welding of the components was developed in this study.

ITR/P1-46

Characterization of Chinese Beryllium as the Candidate Armour Material for ITER First Wall

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Beryllium has been selected as the armor material of ITER first wall blankets. Up to now, only S-65C beryllium made by Brush Wellman Inc. USA, was considered the reference material. China and Russia are the participant members of ITER organization and will share the manufacturing of ITER-FW panels. Therefore beryllium for ITER candidate material has been being developed in both countries. Due to the policy of ITER international organization (IO), candidate ITER materials must be qualified to satisfy the requirements of ITER. As to beryllium materials, an equivalent performance to S-65C beryllium is crucial. In order to demonstrate the damage behaviors of different beryllium grades under ITER operation conditions, two Be/CuCrZr alloy mockups were prepared by means of hot isostatic press (HIP) in which Chinese beryllium, Russian beryllium and S-65C beryllium are included. The Be/CuCrZr mockups were exposed to a high energetic electron beam in the electron beam facility JUDITH at Forschungszentrum Juelich to simulate the transient load condition resulted from plasma disruption, ELMs and vertical displacement vents (VDEs), as well as the thermal fatigue caused by the normal cycling operation of ITER. In this paper, the manufacturing processes and main thermo-mechanical properties of Chinese beryllium are briefly introduced, and qualification test results of the Be/CuCrZr mockups are presented. An equivalent high heat loading performance to S-65C is identified.

ITR/P1-47

Status of Design and R&D for ITER Sector Sub-Assembly Tools

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The ITER (International Thermonuclear Experimental Reactor) tokamak assembly tools are purpose-built tools to complete the ITER tokamak machine which includes the cryostat and the components contained therein. Based on the design description document prepared by the ITER organization, Korea has carried out the conceptual design of assembly tools. The ITER tokamak device is composed of 9 vacuum vessel (VV)/toroidal field coils (TFCs)/vacuum vessel thermal shields (VVTS) 40° sectors. Each VV/TFCs/VVTS 40° sector is made up of one 40° VV, two 20° TFCs and associated VVTS segments. The 40° sectors are sub-assembled at assembly hall respectively and then 9 sectors which sub-assembled at assembly hall are finally assembled at tokamak hall. The 40° sector assembly tools are classified into 2 groups. One group is the sub-assembly tools including upending tool, lifting tool, sub-assembly tool, VV supports and bracing tools used at assembly hall and the other group is the in-pit assembly tools that include lifting tools, radial beams, central column and supports. The process of the ITER sector sub-assembly at assembly hall and status of research and development are described in this paper.

ITR/P1-48

Status of the Korean R&D Program on the ITER Coil Power Converters

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Korean Domestic Agency has been being performing 2-staged converter R&D to establish the technical basis for the procurement package of ITER AC/DC converter. The first phase R&D demonstrated the main features of ITER converter such as the current sharing of 1.28 and FSC (Fault Suppression Capability) at the peak current level of 175 kA with a 1/6-scaled ITER VS converter. The second phase R&D is to build 1/2-scaled ITER converter that realize the full integration of all assemblies, by which the self-supporting aluminum structure that is safe from severe electromagnetic stress under high fault current level of 300 kA will be engineered and verified.

ITR/P1-49

Design and R&D for ITER Feeder System

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ITER tokamak device is currently composed of 31 feeders (26 for magnet system, 3 for structure cooling and 2 for instrumentation) which transfer the power, supercritical Helium and instrumentation wires to the magnets and structures. The feature of the feeders system is that they have many interfaces to other components. So far, TF feeder system engineering design has been finished. R&D work are performed based on the TF feeder design. For feeder system, the aluminum are used for TS of CTB and SBB; the new HTS current leads have excellent performance; many new design are presented for large LF and heat loads. These will speed up and guarantee the feeder system performance definitely in the future. The design and R&D work are described in this paper.

ITR/P1-50

Technology Development for the Manufacture of Nb3Sn Conductors for ITER Toroidal Field coils

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Japan Atomic Energy Agency is procuring the Nb3Sn superconductors for Toroidal Field (TF) Coils under the ITER project. Because manufacturing amount of Nb3Sn strands is quite large compared with the past experience and required superconducting performance is higher than that of the model coils which have been fabricated and tested in the ITER-EDA, quality control technique is very important for the manufacture of the strands. About 40 tons of Nb3Sn strands were completed successfully under the severe quality control and these strands meet the ITER specification. Sophisticated control technique is also required for the jacketing, in order to fabricate the conductors with the precise outer diameter and without leakage at welding part. For example, the inspection technique by using laser for the inner surface of welding part was developed.

Cu dummy conductor with full length (760 m) has been fabricated successfully and all jacketing technology was confirmed through this fabrication. The manufacturing procedure specifications, which satisfy the ITER requirements, were established. The fabrication of the Nb3Sn conductor for TF coils will start in March 2010.

ITR/P1-51

Radiation Resistant Insulation Systems for the ITER Toroidal Field Coils

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Insulation systems for fusion magnets have to operate under intense radiation, mainly fast neutrons and gamma-radiation. At the ITER design fluence of $1\times10^{22}/\mathrm{m}^2$ ($E > 0.1$ MeV), conventional epoxy resins, which were used, e.g., for the ITER TF model coil, start to disintegrate. As a consequence, research programs were launched under EFDA, F4E and ITER contracts in order to find radiation harder resins as well as to investigate the properties of cyanate esters (CE) and CE/epoxy blends. The CE content was varied in the range between 100% and 20%.

Test materials, consisting of wrapped R-glass fiber / polyimide tapes and VPI impregnated with various resins, were produced by European industry and tested at 77K. Tensile and short beam shear tests were carried out to characterize the mechanical properties prior to and after irradiation to fast neutron fluences of up to $5\times10^{22}/\mathrm{m}^2$ ($E > 0.1$ MeV) in the TRIGA reactor (Vienna).

It turned out that these CE based insulation systems offer excellent properties at the ITER design fluence including an adequate safety margin, especially when a mixing ratio of 40% CE and 60% epoxy resin is employed. Slight modifications of the resin, mainly regarding an extension of the pot-life to more than 100 hours, were made in view of the dimensions and the complexity of the ITER TF coils. The corresponding results confirm their excellent performance and will be reported.

Radiation resistant materials have also been developed by US and Japanese industry. The qualification of these insulation systems is currently under way and comprises 40:60 CE blends from different suppliers (Huntsman, CTD). The test materials were fabricated under the same conditions in order to arrive at a comparable data base. The results will be presented at the conference.

ITR/P1-52

R&D of the ITER Correction Coil Magnet System

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This paper is focused on the analysis and R&D of the ITER Correction Coils (CCs). Eighteen superconducting coils are used to compensate field errors arising from misalignment of the coils and winding deviations from the nominal shape as a result of fabrication tolerances, joints, leads, busbars and assembly tolerance. There are 6 top CCs (TCC), 6 bottom CCs (BCC), and 6 side CCs (SCC), arranged toroidally around the machine inside the PF coils and mounted on the TF coils.

The conductor used in the CCs is NbTi/Cu cable-in conduit (CIC) conductor and its operating current is 10 kA. The primary FE analyses about the magnetic field, the structure and the thermal analysis for the CCs have been made and the design of the CCs has been optimized during the last two years. The R&D programs on the manufacturing of the CCs are in progress, which focused on the pre-bending of the conductors, winding machine, short sample of VPI, welding process of the coil case, and some other key technologies.

ITR/P1-53

Recent Development and Qualification of Materials for ITER

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The material development together with the important qualification of each individual material and joint are continuously providing new data. Results are provided for ITER components contributing to the First Wall/Blanket, the Divertor and the Vacuum Vessel. The work has been organized by EFDA and F4E together with EURATOM associations and industry. Assessment of data including the interfaces to the primary heat transfer systems under ITER operational conditions is highlighted in the presented paper. Historically it is well known that corrosion has caused high costs to nuclear power plants and efforts aimed at avoiding similar situations for ITER have been made or are currently ongoing. Due to the complex geometry and component assembly it is of great importance to understand the corrosion aspects. The initial activity was aimed at identifying possible weak points, such as crevices, joints and welds, within the primary heat transfer systems. Taking into account highlighted corrosion issues along with results from previous and presently ongoing work it is possible to make an assessment of fundamental corrosion parameters. The assessment is done by analysis and simulation, which is followed up by experiments under ITER relevant operational conditions. The drying sequence of the

vacuum vessel using hot nitrogen and hot dry sTeam was simulated using the computer code MELCOR.

Radiolysis calculations were reviewed in a task to determine how the corrosion potential (ECP) changes during the plasma burning cycles. Even if the corrosion effects are found to emerge slowly it is important to follow up the margins against corrosion during the expected lifetime of ITER. The knowledge gathered from these studies also enables preventive actions to be available during operation. A new grade of carbon fibre composite (CFC), DMS 814 MEGGAGARD, was developed by Meggitt UK. The CFC is used as a plasma facing material for the Divertor and it was concluded that this grade can be manufactured to meet ITER requirements. The ITER Organization has identified a lack of materials data concerning the copper alloy CuCrZr and stainless steel 316L(N)-IG and especially the CuCrZr/316L(N)-IG joints. Several studies have been launched as an effort to increase the available materials data. The studies on the CuCrZr/316L joints are contributing to the development of the First Wall.

ITR/P1-54

Fabrication and Testing of the EU FW Qualification Mock-up

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ITER requires the Domestic Agencies (DA's) to be pre-qualified prior to participate in the supply of critical components, among which the Blanket First Wall (FW). Phase 1 of the qualification campaign for the ITER FW consisted of the fabrication and testing of 2 small-scale mock-ups (SSMU) – 80×240 mm – to demonstrate the ability of the selected fabrication technology to resist to the expected thermal loads. In particular, the EU-DA has manufactured and tested 2 SSMU's.

Manufacturing has taken place at CEA, in Grenoble. The manufacturing route selected for these FWQ mock-ups is based on the Hot Isostatic Pressing (HIPping) of solid parts. The grooved CuCrZr plates and the 316L(N)-IG SS tubes were joined together with one single HIP cycle at 1040°C and 140 MPa for 2 hours. After HIP, the mock-ups underwent a solution annealing heat treatment at 980°C for 1 hour with a cooling rate higher than 50°C/min. Then the Beryllium (Be) tiles were joined by HIPping at 580°C and 140 MPa for 2 hours. The mock-ups were then cleaned and ultrasonic tested to check the soundness of the Be/CuCrZr joints.

Testing has been done in different places: in Sandia N.L. in the USA where the mock-up underwent 12 000 cycles at 0.88 MW/m^2 as fatigue testing. At the Nuclear Research Institute (NRI Rez, plc) in the Czech Republic the mock-up underwent another 12 000 cycles at 0.625 MW/m^2 as fatigue testing. Eventually, the 1000 cycles of the MARFE tests at 1.75 MW/m^2 were successfully achieved at Forschungszentrum Juelich (FzJ) in Germany. There, the mock-up was further tested 100 cycles up to 2.75 MW/m^2 fully validating HIP as a robust bonding technology for the considered FW materials. Ultrasonic testing of the mock-up was performed prior and after each high heat flux test campaign.

ITR/P1-55

R&D on 52 kA HTS Trial Current Lead for ITER

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CASIPP has the experience of design and fabrication of 16kA HTSCLS those are successfully used in EAST Tokamak. The aim of the 52 kA lead trial is to develop 45/52 kA CS/PF HTS current lead prototype for the ITER project. Different from former HTSCL structure 52 kA HTSCL current lead not only consists of conventional helium cooled heat exchanger, HTS module using InnoST tapes and NbTi low temperature superconductor (LTS) module but also a helium cooled copper transition and a binary shunt made of S.S and PSnCu. 52 kA HTS current lead was tested in the EAST Tokamak facility at ASIPP. Some test results are also presented and analyzed on the rear of the paper.

SEE

Safety, Environmental and Economic Aspects of Fusion

SEE — Oral

SEE/1-1Ra

Fusion Energy and Climate Change

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Analysis of the energy modeling literature suggests that of the 12,000 GWe electric consumption projected in 2100, 30% may need to be provided by nuclear energy. The annual fueling for a 3600 GWe fleet of fast reactors would be enough to produce $\sim$ 1 million first-generation nuclear weapons per year. If fusion begins to enter commercial operation in 2050, and grows at a rate $\leq$ 0.84%/yr of the world electric market (fission grew in 1975 – 1990 at 1.2%/year of the then electric market) it can provide the needed electrical energy with much lower proliferation risk.

SEE/1-1Rb

The Potential Role for Fusion Power in Future Energy Markets

Mühlich, P.[2], Hamacher, T.[2], Ward, D.[4], Han, W.[4], Eherer, C.[5], Silva, C.A.S.[7], Gracceva, F.[3], Ciorba, U.[3], Biberacher, M.[5], Grohnheit, P.E.[6], Lechon, Y.[1], Pina, A.[7], Cabal Cuesta, H.[1], Hartmann, T.[2], and Tosato, G.C.[4]

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In order to explore the potential role for fusion in a future energy market, and clarify the conditions under which fusion may be important in different world regions, a global energy scenario model, based on the model generator TIMES supplied by the International Energy Agency, has been developed. The model covers the whole of this century and includes fusion technologies. Results are reported here.

SEE/1-1Rc

Environmental and Economic Assessments of Magnetic and Inertial Confinement Fusion Reactors

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Global warming due to rapid greenhouse gas (GHG) emission is one of the present-day crucial problems, and fusion reactors are expected as abundant electric power generation systems to reduce GHG emission amounts. To search for environment-friendly and economic fusion reactor system, comparative system studies have been done for several magnetic confinement fusion (MCF) reactors, and recently extended to include inertial confinement fusion (ICF) reactors.

At first, economic assessment models for MCF reactors including tokamak (TR), helical (HR) and spherical tokamak (ST) systems are described. These MCF reactor designs strongly depend on achievable plasma beta value and magnetic field strength. In the system code we confirmed the advantage of high-beta TR designs in cost of electricity (COE). After wide parameter scans, we obtained the new COE scaling formulas for MCF reactors as functions of electric power ($1-3$ GW), plant availability ($0.65-0.85$), normalized beta ($3-5$) or averaged beta ($3-5\%$), maximum magnetic field strength ($10-16$ T), thermal efficiency ($0.37-0.59$) and operation year ($20-40$ Year).

In the case of ICF reactors, fast ignition concept is adopted here. The target gain and driver repetition rate are evaluated assuming driver energy, driver efficiency (~ 0.075), compression efficiency (~ 0.05) and heating efficiency (~ 0.10), which critically determine the fusion core system.

The life-cycle CO_2 emissions equivalently including methane gas are also evaluated using the input-output table method. High power plant assessment and carbon tax effect on COE are also evaluated, and fusion power plants are compared with other electric power plants. The fusion reactors emit less GHG than fossil fuel thermal power plant. In comparison with fission reactors, the fusion has a disadvantage in COE, but has an advantage in GHG emission. When the carbon tax of around 3,000 yen/t-CO2equi is introduced, the COE of fusion reactor might be at the same level on that of coal-fired electric power plant without CCS (Carbon dioxide Capture and Storage) system and 1.5 times lower than that of oil-fired electric power plant.

In thess assessment studies, we clarify new scaling formulas for GHG emissions with respect to key design parameters, which might be important for making a strategy of fusion research development.

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